PSEG NUCLEAR LLC
DOCKET 50-354
HOPE CREEK GENERATING STATION
RENEWED FACILITY OPERATING LICENSE

Renewed License No. NPF-57

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

   A. The application for a renewed license filed by the PSEG Nuclear LLC (the licensee), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Hope Creek Generating Station (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-120 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D below);

   D. There is reasonable assurance: (i) that the activities authorized by this renewed operating license 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's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D below);

   E. PSEG Nuclear LLC is technically qualified to engage in the activities authorized by this renewed license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

   H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Renewed Facility Operating License No. NPF-57, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied;

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70;

J. The receipt, production, possession, transfer, and use of Cobalt-60 as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Part 30; and

K. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, "Definitions," for the facility, and that any changes made to the facility's current licensing basis to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

2. Based on the foregoing findings and approval by the Nuclear Regulatory Commission at a meeting on July 21, 1986, the License for Fuel Loading and Low-Power Testing, License No. NPF-50, issued on April 11, 1986, is superseded by Renewed Facility Operating License No. NPF-57 hereby issued to PSEG Nuclear LLC (the licensee), to read as follows:

A. This renewed license applies to the Hope Creek Generating Station, a boiling water nuclear reactor, and associated equipment (the facility) owned by PSEG Nuclear LLC. The facility is located on the licensee's site on the east bank of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey. The facility is located approximately eight miles southwest of Salem, New Jersey and is described in the PSEG Nuclear LLC Final Safety Analysis Report, as supplemented and amended, and in the Environmental Report, as supplemented and amended.

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

(1) PSEG Nuclear LLC, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the above designated location in Salem County, New Jersey, in accordance with the procedures and limitations set forth in this renewed license;

(2) Deleted

(3) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for

reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time 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 required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation.

(7) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3840 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 192, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

(4) Inservice Inspection (Section 6.6, SER; Sections 5.2.4.3 and 6.6.3, SSER No. 5)

a. PSE&G shall submit an inservice inspection program in accordance with 10 CFR 50.55a(g)(4) for staff review by October 11, 1986.

b. Pursuant to 10 CFR 50.55a(a)(3) and for the reasons set forth in Sections 5.2.4.3 and 6.6.3 of SSER No. 5, the relief identified in the PSE&G submittal dated November 18, 1985, as revised by the submittal dated January 20, 1986, requesting relief from certain requirements of 10 CFR 50.55a(g) for the preservice inspection program, is granted.

(5) Solid State Logic Modules

PSEG Nuclear LLC shall continue, for the life of the plant, a reliability program to monitor the performance of the Bailey 862 SSLMs installed at Hope Creek Generating Station. This program should obtain reliability data, failure characteristics, and root cause of failure of both safety-related and non-safety-related Bailey 862 SSLMs. The results of the reliability program shall be maintained on-site and made available to the NRC upon request.

(6) Fuel Storage and Handling (Section 9.1, SSER No. 5)

a. No more than a total of three (3) fuel assemblies shall be out of approved shipping containers, NRC-approved dry spent fuel storage systems, fuel assembly storage racks or the reactor at any one time.

b. The above three (3) fuel assemblies as a group shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and the storage rack array.

c. Fresh Fuel assemblies, when stored in their shipping containers, shall be stacked no more than three (3) containers high.

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* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(7) Fire Protection (Section 9.5.1.8, SSER No. 5; Section 9.5.1, SSER No. 6)

PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 15 and as described in its submittal dated May 13, 1986, and as approved in the SER dated October 1984 (and Supplements 1 through 6) subject to the following provision:

PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(8) Solid Waste Process Control Program (Section 11.4.2, SER; Section 11.4, SSER No. 4)

PSEG Nuclear shall obtain NRC approval of the Class B and C solid waste process control program prior to processing Class B and C solid wastes.

(9) Emergency Planning (Section 13.3, SSER No. 5)

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR 50.54(s)(2) will apply.

(10) Initial Startup Test Program (Section 14, SSER No. 5)

Any changes to the Initial Startup Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(11) Partial Feedwater Heating (Section 15.1, SER; Section 15.1, SSER No. 5; Section 15.1, SSER No. 6)

The facility shall not be operated with a rated thermal power feedwater temperature less than 329.6°F for the purpose of extending the normal fuel cycle.

(12) Detailed Control Room Design Review (Section 18.1, SSER No. 5)

a. PSE&G shall submit for staff review Detailed Control Room Design Review Summary Reports II and III on a schedule consistent with, and with contents as specified in, its letter of January 9, 1986.

b. Prior to exceeding five percent power, PSE&G shall provide temporary zone markings on safety-related instruments in the control room.

(13) <u>Safety Parameter Display System (Section 18.2, SSER No. 5)</u>

Prior to the earlier of 90 days after restart from the first refueling outage or July 12, 1988, PSE&G shall add the following parameters to the SPDS and have them operational:

a. Primary containment radiation
b. Primary containment isolation status
c. Combustible gas concentration in primary containment
d. Source range neutron flux

(14) <u>Additional Conditions</u>

The Additional Conditions contained in Appendix C, as revised through Amendment No. 135, are hereby incorporated into this renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Additional Conditions.

(15) PSE&G to PSEG Nuclear LLC License Transfer Conditions

a. PSEG Nuclear LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application, the requirements of the Order Approving Transfer of License and Conforming Amendment, dated February 16, 2000, and the related Safety Evaluation dated February 16, 2000.

b. The decommissioning trust agreement shall provide that:

1) The use of assets in both the qualified and non-qualified funds shall be limited to expenses related to decommissioning of the unit as defined by the NRC in its regulations and issuances, and as provided in the unit's renewed license and any amendments thereto. However, upon completion of decommissioning, as defined above, the assets may be used for any purpose authorized by law.

2) Investments in the securities or other obligations of PSE&G or affiliates thereof, or their successors or assigns,

shall be prohibited. In addition, except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants shall be prohibited.

3) No disbursements or payments from the trust shall be made by the trustee until the trustee has first given the NRC 30 days notice of the payment. In addition, no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the Director, Office of Nuclear Reactor Regulation.

4) The trust agreement shall not be modified in any material respect without prior written notification to the Director, Office of Nuclear Reactor Regulation.

5) The trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(3) of the Federal Energy Regulatory Commission's regulations.

c. PSEG Nuclear LLC shall not take any action that would cause PSEG Power LLC or its parent companies to void, cancel, or diminish the commitment to fund an extended plant shutdown as represented in the application for approval of the transfer of this license from PSE&G to PSEG Nuclear LLC.

(16) Mitigation Strategy

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread

4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

(17) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(18) Upon implementation of Amendment No. 173 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by Surveillance Requirement 4.7.2.2.a, in accordance with TS 6.16.c.(i), the assessment of CRE habitability as required by Specification 6.16.c.(ii), and the measurement of CRE pressure as required by Specification 6.16.d, shall be considered met. Following implementation:

a. The first performance of Surveillance Requirement 4.7.2.2.a, in accordance with Specification 6.16.c.(i), shall be within the specified frequency of 6 years, plus the 18 month allowance of Surveillance Requirement 4.0.2, as measured from July 29, 2001, the date of the most recent successful tracer gas test, as stated in the December 9, 2003 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

b. The first performance of the periodic assessment of CRE habitability, Specification 6.16.c(ii), shall be 3 years, plus the 9 month allowance of Surveillance Requirement 4.0.2, as measured from July 29, 2001, the date of the most recent successful tracer gas test, as stated in the December 9, 2003 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

c. The first performance of the periodic measurement of CRE pressure, Specification 6.16.d, shall be within 18 months, plus the 138 days allowed by Surveillance Requirement 4.0.2, as measured from April 5, 2006, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.

(19) Leak rate tests required by Surveillance Requirement 4.6.1.2.a and 4.6.1.2.h to be performed in accordance with the Primary Containment Leakage Rate Testing Program are not required to be performed until their next scheduled performance, which is due at the end of the first test interval that begins on the date the test was last performed prior to implementation of Amendment No. 174.

(20) Top Guide Beams

Until there is more detailed guidance regarding the inspections of the top guide beams or the issue is resolved by the BWRVIP generically, the following license condition applies to Hope Creek to preclude the loss of the component's intended function:

Enhanced visual testing (EVT-1) of the top guide grid beams will be performed in accordance with GE SIL 554 following the sample selection and inspection frequency of BWRVIP-47 for CRD guide tubes. That is, inspections will be performed on 5 percent of the population within six years, and 10 percent of the total population of cells within twelve years. The sample locations selected for examination will be in areas that are exposed to the highest fluence. This inspection plan will be implemented beginning with the first RFO following EPU operation.

(21) Vibration Acceptance Criteria for SRVs

PSEG Nuclear LLC shall provide the Level 1 main steam safety relief valve vibration acceptance criteria to the NRC staff prior to increasing power above 3339 MWt.

(22) Steam Dryer

This license condition provides for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the stream dryer).

1. The following requirements are placed on initial operation of the facility at power levels above 3339 MWt to 3840 MWt for the power ascension:

   a. PSEG Nuclear LLC shall monitor hourly the main steam line (MSL) strain gage data during power ascension above 3339 MWt for increasing pressure fluctuations in the steam lines.

   b. PSEG Nuclear LLC shall hold the facility at 105 percent and 110 percent of 3339 MWt to collect data from the MSL strain gages required by Condition 1.a, conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data; shall submit the evaluation to the NRC staff upon completion of the evaluation; and shall not increase power above each hold point until 96 hours after submitted to the NRC.

   c. If any frequency peak from the MSL strain gage data exceeds any of the Level 1 limit curves, PSEG Nuclear LLC shall return the facility to a lower power level at which the limit curve is not exceeded. PSEG Nuclear shall resolve the uncertainties in the steam dryer analysis, evaluate the continued structural integrity of the steam dryer, and submit that evaluation to the NRC staff.

   d. In addition to evaluating the MSL strain gage data, PSEG Nuclear LLC shall monitor reactor pressure vessel water level instrumentation and MSL piping accelerometers on an hourly basis during power ascension above 3339 MWt. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gage instrumentation data (including consideration of the EPU bump-up factor), PSEG Nuclear LLC shall stop power ascension, evaluate the continued structural integrity of the steam dryer, and submit that evaluation to the NRC staff.

2. PSEG Nuclear LLC shall implement the following actions for the initial power ascension at power levels above 3339 MWt to 3840 MWt:

    a. In the event that acoustic signals are identified that challenge the limit curves during power ascension above 3339 MWt, PSEG Nuclear LLC shall evaluate dryer loads and re-establish the limit curves based on the new strain gage data, and shall perform a frequency-specific assessment of ACM uncertainty at the acoustic signal frequency including application of 65 percent bias error and 10 percent uncertainty to all the SRV acoustic resonances.

    b. After reaching 111.5 percent of 3339 MWt, PSEG Nuclear LLC shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and reestablish the limit curves with the updated ACM load definition, which will be submitted to the NRC staff.

    c. After reaching 115 percent of 3339 MWt, PSEG Nuclear LLC shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the limit curves with the updated ACM load definition, which will be submitted to the NRC staff.

    d. During power ascension above 3339 MWt, if an engineering evaluation is required because a Level 1 acceptance criterion is exceeded, PSEG Nuclear LLC shall perform the structural analysis to address frequency uncertainties up to ±10 percent and assure that peak responses that fall within this uncertainty band are addressed.

    e. PSEG Nuclear LLC shall revise plant procedures to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with BWRVIP-139; and to identify the NRC Project Manager for the facility as the point of contact for providing power ascension testing information during power ascension.

f. PSEG Nuclear LLC shall submit the final EPU steam dryer load definition for the facility to the NRC staff upon completion of the power ascension test program.

g. PSEG Nuclear LLC shall submit the flow-induced vibration related portions of the EPU startup test procedure to the NRC staff, including methodology for updating the limit curves, prior to initial power ascension above 3339 MWt.

3. PSEG Nuclear LLC shall prepare the EPU startup test procedure to include:

a. the stress limit curves to be applied for evaluating steam dryer performance;

b. specific hold points and their duration during EPU power ascension;

c. activities to be accomplished during hold points;

d. plant parameters to be monitored;

e. inspections and walk downs to be conducted for steam, FW, and condensate systems and components during the hold points;

f. methods to be used to trend plant parameters;

g. acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections;

h. actions to be taken if acceptance criteria are not satisfied; and

i. verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above 3339 MWt.

PSEG Nuclear LLC shall provide the related EPU startup test procedure sections to the NRC staff prior to increasing power above 3339 MWt.

4. The following key attributes of the program for verifying the continued structural integrity of the steam dryer shall not be made less restrictive without prior NRC approval:

a. During initial power ascension testing above CLTP, each test plateau increment shall be approximately 5 percent of 3339 MWt;

b. Level 1 performance criteria; and

c. The methodology for establishing the stress spectra used for the Level 1 and Level 2 performance criteria.

Changes to other aspects of the program for verifying the continued structural integrity of the steam dryer may be made in accordance with the guidance of NEI 99-04.

5. During the first scheduled refueling outage after Cycle 15 and during the first two scheduled refueling outages after reaching full EPU conditions, a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer in accordance with BWRVIP-139 inspection guidelines.

6. The results of the visual inspections of the steam dryer shall be reported to the NRC staff within 90 days following startup from the respective refueling outage. The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report within 60 days following the completion of all Cycle 15 power ascension testing. A supplement shall be submitted within 60 days following the completion of all EPU power ascension testing.

(23) Irradiated GE14i fuel bundles shall be stored at least four feet from the wall of the Spent Fuel Pool.

(24) PSEG Nuclear LLC may make changes to the programs and activities described in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(25) Appendix A of NUREG-2102, “Safety Evaluation Report Related to the License Renewal of Hope Creek Generating Station,” dated June 2011, and the licensee's UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on May 19, 2011, describes certain future programs and activities to be completed before the period of extended operation. PSEG Nuclear LLC shall complete these activities no later than April 11, 2026, and shall notify the NRC in writing when implementation of these activities is complete.

(26) The licensee will establish drainage capability from the bottom of the drywell air gap on or before June 30, 2015. The licensee will divide the drywell air gap into four approximately equal quadrants. Drainage consists of one drain in each quadrant for a total of four drains. Each drain will be open at the bottom of the drywell air gap and be capable of draining water from the air gap.

Until drainage is established from all four quadrants, the licensee will perform the following actions each refueling outage:

a. Perform boroscope examination of the bottom of the drywell air gap through penetrations located at elevation 93'-0" in four quadrants, 90 degrees apart. The personnel performing the boroscope examination shall be certified as VT-1 inspectors in accordance with ASME Section XI, Subsection IWA-2300, requirements. The examiners will look for signs of water accumulation and drywell shell corrosion. Adverse conditions will be documented and addressed in the corrective action program.

b. Perform ultrasonic thickness (UT) measurements of the drywell shell between elevations 86'-11" (floor of the drywell concrete) and 93'-0" (bottom of penetration J13) below penetration J13 area. In addition, UT measurements shall be performed around the full 360 degree circumference of the drywell between elevations 86'-11" and 88'-0" (underside of the torus down comer vent piping penetrations). The results of the UT measurements shall be used to establish a corrosion rate and demonstrate that the effects of aging will be adequately managed such that the drywell can perform its intended function until April 11, 2046. Evidence of drywell shell degradation will be documented and addressed in the corrective action program.

c. Monitor penetration sleeve J13 daily for water leakage when the reactor cavity is flooded up. In addition, perform a walkdown of the torus room to detect any leakage from other drywell penetrations. These actions shall continue until corrective actions are taken to prevent leakage through J13.

d. Within 90 days of completion of each refueling outage, submit a report to the NRC staff in accordance with 10 CFR 50.4 summarizing the results from the boroscope examinations, UT measurements, leakage detected from penetrations, and if appropriate, corrective action.

(27) After drainage has been established from the bottom of the air gap in all four quadrants, the licensee will:

a. Submit a report to the NRC staff in accordance with 10 CFR 50.4 describing the final drain line configuration and summarizing the testing results that demonstrate drainage has been established for all four quadrants.

b. Monitor penetration sleeve J13 daily for water leakage when the reactor cavity is flooded up. In addition, perform a walkdown of the torus room to detect any leakage from other drywell penetrations. These actions shall continue until corrective actions are taken to prevent leakage through J13 or through the four air gap drains.

c. Perform UT measurements of the drywell shell between elevation 86'-11" (floor of the drywell concrete) and elevation 93'-0" (bottom of penetration J13) below penetration J13 area during the next three refueling outages. In addition, UT measurements shall be performed around the full 360 degree circumference of the drywell between elevations 86'-11" and 88'-0" (underside of the torus down comer vent piping penetrations). The results of the UT measurements will be used to identify drywell surfaces requiring augmented inspections in accordance with IWE requirements for the period of extended operation, establish a corrosion rate, and demonstrate that the effects of aging will be adequately managed such that the drywell can perform its intended function until April 11, 2046. Within 90 days of completion of each refueling outage, submit a report to the NRC staff in accordance with 10 CFR 50.4 summarizing the results from the UT measurements and if appropriate, corrective action.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. An exemption from the criticality alarm requirements of 10 CFR 70.24 was granted in Special Nuclear Material License No. 1953, dated August 21, 1985. This exemption is described in Section 9.1 of Supplement No. 5 to the SER. This previously granted exemption is continued in this renewed operating license. An exemption from certain requirements of Appendix A to 10 CFR Part 50, is described in Supplement No. 5 to the SER. This exemption is a schedular exemption to the requirements of General Design Criterion 64, permitting delaying functionality of the Turbine Building Circulating Water System-Radiation Monitoring System until 5 percent power for local indication, and until 120 days after fuel load for control room indication (Appendix R of SSER 5). Exemptions from certain requirements of Appendix J to 10 CFR Part 50, are described in Supplement No. 5 to the SER. These include an exemption from the requirement of Appendix J, exempting main steam isolation valve leak-rate testing at 1.10 Pa (Section 6.2.6 of SSER 5); an exemption from Appendix J, exempting Type C testing on traversing incore probe system shear valves (Section 6.2.6 of SSER 5); an exemption from Appendix J, exempting Type C testing for instrument lines and lines containing excess flow check valves (Section 6.2.6 of SSER 5); and an exemption from Appendix J, exempting Type C testing of thermal relief valves (Section 6.2.6 of SSER 5). These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. These exemptions are hereby granted. The special circumstances regarding each exemption are identified in the referenced section of the safety evaluation report and the supplements thereto. These exemptions are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, submitted by letter dated May 19, 2006 are entitled: "Salem-Hope Creek Nuclear Generating Station Security Training and Qualification Plan," and "Salem-Hope Creek Nuclear Generating Station Security Contingency Plan." The plans contain Safeguards Information protected under 10 CFR 73.21.

PSEG Nuclear LLC shall fully implement and maintain in effect all provisions of the Commission-approved Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Salem-Hope Creek CSP was approved by License Amendment No. 189 as supplemented by a change approved by License Amendment No.192.

F. DELETED

G. 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.

H. This renewed license is effective as of the date of issuance and shall expire at midnight on April 11, 2046.

FOR THE NUCLEAR REGULATORY COMMISSION

- original signed by E. J. Leeds -

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A - Technical Specifications (NUREG-1202)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional Conditions

Date of Issuance: July 20, 2011

NUREG-1202

Technical Specifications
Hope Creek Generating Station

Docket No. 50-354

Appendix "A" to
License No. NPF-57

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

July 1986

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DEFINITIONS

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

DESIGN FEATURES

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

1.2 DELETED

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at that height.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. 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 so that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

CORE ALTERATION

1.7 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement), and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

1.8 DELETED

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC-approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.14 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

1.15 DELETED

1.16 DELETED

FREQUENCY NOTATION

1.17 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

1.23 DELETED

MEMBER(S) OF THE PUBLIC

1.24 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, it contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.25 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFF-GAS RADWASTE TREATMENT SYSTEM

1.26 An OFF-GAS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting reactor coolant system offgases from the main condenser evacuation system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATIONAL MANUAL (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/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Annual Radioactive Effluent Release Report required by Specifications 6.9.1.6 and 6.9.1.7.

OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packing of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3840 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve or damper, as applicable secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The filtration, recirculation and ventilation system is in compliance with the requirements of Specification 3.6.5.3.

d. For double door arrangements, at least one door in each access to the secondary containment is closed.

e. For single door arrangements, the door in each access to the secondary containment is closed, except for normal entry and exit..

f. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

g. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.40 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.41 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled, by the licensee.

SOLIDIFICATION

1.42 Not used.

SOURCE CHECK

1.43 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

SPIRAL RELOAD

1.44 A SPIRAL RELOAD is a core loading methodology employed to refuel the core after a complete core unload. During a SPIRAL RELOAD the fuel is to be loaded into individual control cells (four bundles surrounding a control blade) in a spiral fashion centered on an SRM moving outward. Before initiating a SPIRAL RELOAD, up to four bundles may be loaded in the four bundle locations immediately surrounding each of the four SRMs to obtain the required channel count rate.

1.45 A SPIRAL UNLOAD is a core unloading methodology employed to defuel when the complete core is to be unloaded. The core unload is performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for the four bundles around each of the four SRMs. When sixteen or less fuel bundles are in the core, four around each of the four SRMs, there is no need to maintain the required channel count rate.

STAGGERED TEST BASIS

1.46 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.47 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.48 The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two separate time intervals: a) time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established, and b) the time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. Either response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.49 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.50 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.51 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.52 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
Z	During startup, prior to exceeding 30% of RATED THERMAL POWER, if not performed within the previous 7 days
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown#,***	> 200°F
4. COLD SHUTDOWN	Shutdown#,##,***	≤ 200°F+
5. REFUELING*	Shutdown or Refuel**,#	≤ 140°F

#The reactor mode switch may be placed in the Run, Startup/Hot Standby, or Refuel position to test the switch interlock functions and related instrumentation provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff. If the reactor mode switch is placed in the Refuel position, the one-rod-out interlock shall be OPERABLE.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

+See Special Test Exception 3.10.8.

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 24% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 24% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow:

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be ≥ 1.08 for two recirculation loop operation and shall be ≥ 1.10 for single recirculation loop operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow and the MCPR below the values for the fuel stated in LCO 2.1.2, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 14% of RATED THERMAL POWER	≤ 19% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.57(w-Δw) + 58%** with a maximum of	≤ 0.57(w-Δw) + 61%** with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed

*See Bases Figure B 3/4 3-1.

**The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w). Δw is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. Δw = 0 for two recirculation loop operation. Δw = 9% for single recirculation loop operation.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS
(continued)

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. This item intentionally blank		
7. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	Elevation 110' 10.5"	Elevation 111' 0.5"
b. Level Transmitter/Trip Unit	Elevation 110' 10.5"*	Elevation 111' 0.5"*
9. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ 530 psig	$\geq$ 465 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

*80.5" above instrument zero EL 104' 2" for Level Transmitter/Trip Unit A&B (South Header) 83.25" above instrument zero EL 103' 11.25" for Level Transmitter/Trip Unit C&D (North Header)

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, and except as provided in LCO 3.0.8.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL CONDITIONS 4 or 5.

3.0.4 When an LCO is not met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time; or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LIMITING CONDITION FOR OPERATION (Continued)

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 testing required to demonstrate OPERABILITY.

3.0.6 Not used.

3.0.7 Not used.

3.0.8 Inoperability of Snubbers

When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified frequency shall be a failure to meet the Limiting Condition for Operation, except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within its specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever 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 Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. In accordance with the Surveillance Frequency Control Program during POWER OPERATION.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. Immediately:

      a) Verify that the inoperable control rod, if withdrawn, meets the stuck control rod separation criteria.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

   2. Within two hours:

      a) Disarm the associated control rod drive.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above:

   1. Within three hours: insert the inoperable withdrawn control rod(s).

   2. Within four hours disarm the associated control rod drive.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods*****:

   1. Within 4 hours, restore compliance with BPWS, or

   2. Within 4 hours, restore control rod(s) to OPERABLE status, or

   3. Within 8 hours, verify control rod drop accident limits are met.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. One or more BPWS groups with four or more inoperable control rods*****, within 4 hours, restore control rod(s) to OPERABLE status.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

e. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

f. With one or more scram discharge volume (SDV) vent or drain lines*** with one valve inoperable, isolate the associated line within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.****

g. With one or more SDV vent or drain lines*** with both valves inoperable, isolate the associated line within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.****

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Verifying each valve to be open,* and

b. Cycling each valve through at least one complete cycle of full travel.

---

* These valves may be closed intermittently for testing under administrative controls.

** May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

*** Separate Action entry is allowed for each SDV vent and drain line.

**** An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

***** Not applicable when THERMAL POWER is greater than 8.6% RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.2 When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. In accordance with the Surveillance Frequency Control Program, and

b. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.3, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE in accordance with the Surveillance Frequency Control Program, by verifying that the drain and vent valves:

    1. Close within 30 seconds after receipt of a signal for control rods to scram, and

    2. Open when the scram signal is reset.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

   1. Declare the control rod(s) with the slow insertion time inoperable

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Verify each control rod scram time from fully withdrawn to notch position 05 is ≤ 7.0 seconds in accordance with Surveillance Requirement 4.1.3.3.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.3 No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.3.3-1, and no more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

Table 3.1.3.3-1

---------- NOTES ----------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3.2, "Control Rod Maximum Scram Insertion Times," for control rods with scram times > 7.0 seconds to notch position 05. These control rods are inoperable in accordance with SR 4.1.3.2 and are not considered "slow."

Notch Position	Scram Times(a)(b) (Seconds) When Reactor Steam Dome Pressure ≥ 800 psig
45	0.52
39	0.86
25	1.91
05	3.44

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With more than 13 OPERABLE control rods exceeding any of the above limits or more than 2 OPERABLE control rods that are "slow" occupy adjacent locations, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3 During single control rod scram time surveillances with the control rod drive pumps isolated from the accumulators:

a. Verify each control rod scram time is within the limits of Table 3.1.3.3-1 with reactor steam dome pressure ≥ 800 psig prior to THERMAL POWER exceeding 40% RATED THERMAL POWER after each reactor shutdown ≥ 120 days.

b. Verify for a representative sample, each tested control rod scram time is within the limits of Table 3.1.3.3-1 with reactor steam dome pressure ≥ 800 psig in accordance with the Surveillance Frequency Control Program.

c. Verify each affected control rod scram time is within the limits of Table 3.1.3.3-1 with any reactor steam dome pressure prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time.

d. Verify each affected control rod scram time is within the limits of Table 3.1.3.3-1 with reactor steam dome pressure ≥ 800 psig prior to THERMAL POWER exceeding 40% RATED THERMAL POWER after fuel movement within the affected core cell AND prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time.

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 Deleted

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

--NOTE--

Separate condition entry is allowed for each control rod

---

a. In OPERATIONAL CONDITIONS 1 or 2:

1. With one control rod scram accumulator inoperable and reactor pressure ≥ 900 psig, within 8 hours,

   a) Restore the inoperable accumulator to OPERABLE status, or

   b) Declare the associated control rod scram time "slow"***, or

   c) Insert the associated control rod, declare the associated control rod inoperable and disarm the associated control valves by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN with the next 12 hours.

2. With two or more control rod scram accumulators inoperable and reactor pressure ≥ 900 psig,

   a) Within 20 minutes of discovery of this condition concurrent with charging water pressure < 940 psig, restore charging water header pressure to ≥ 940 psig otherwise place the mode switch in the shutdown position**, and

   b) Within one hour, declare the associated control rod scram time "slow"***, or

   c) Within one hour insert the associated control rods, declare the associated control rods inoperable and disarm the associated control valves by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

---

* At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

*** Only applicable if the associated control rod scram time was within the limits of Table 3.1.3.3-1 during the last scram time Surveillance. Rods that are already considered "slow" should be declared inoperable and fully inserted.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

3. With one or more control rod scram accumulators inoperable and reactor pressure < 900 psig,

a) Immediately upon discovery of charging water header pressure < 940 psig, verify all control rods associated with inoperable accumulators are fully inserted otherwise place the mode switch in the shutdown position**, and

b) Within one hour insert the associated control rod(s), declare the associated control rod(s) inoperable and disarm the associated control valves either electrically or hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one or more withdrawn control rods inoperable, upon discovery immediately initiate action to fully insert inoperable withdrawn control rods.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed.

---

* At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

LIMITING CONDITION FOR OPERATION

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 and 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   1. If permitted by the RWM, insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      a) Observing any indicated response of the nuclear instrumentation, and
      b) Demonstrating that the control rod will not go to the overtravel position.

   2. If recoupling is not accomplished on the first attempt or, if not permitted by the RWM, then until permitted by the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours either:

   1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or

   2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.3.6 Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within 1 hour:

    1. Determine the position of the control rod by using an alternative method, or:

        a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,
        b) Returning the control rod, by single notch movement, to its original position, and
        c) Verifying no control rod drift alarm at least once per 12 hours, or

    2. Move the control rod to a position with an OPERABLE position indicator, or

    3. When THERMAL POWER is:

        a) Within the preset power level of the RWM, declare the control rod inoperable.
        b) Greater than the preset power level of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

            1) Electrically, or
            2) Hydraulically by closing the drive water and exhaust water isolation valves.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

a. In accordance with the Surveillance Frequency Control Program that the position of each control rod is indicated,

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and

c. That the control rod position indicator corresponds to the control rod position indicated by the "Full Out" position indicator when performing Surveillance Requirement 4.1.3.6.b.

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to 8.6% of RATED THERMAL POWER, minimum allowable low power setpoint.

ACTION:

a. With the RWM inoperable after the first 12 control rods are fully withdrawn, operation may continue provided that control rod movement and compliance with the prescribed control rod pattern is verified by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console.

b. With the RWM inoperable before the first twelve (12) control rods are fully withdrawn, one startup per calendar year may be performed provided that the control rod movement and compliance with the prescribed control rod pattern are verified by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console.

c. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 within 8 hours prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence control rod.

* Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.
# See Special Test Exception 3.10.2.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

SURVEILLANCE REQUIREMENTS (CONTINUED)

b. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

c. In OPERATIONAL CONDITION 1 within one hour after RWM automatic initiation when reducing THERMAL POWER, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

d. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

REACTIVITY CONTROL SYSTEMS

ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

The material originally contained in Section 3/4.1.4.2 was deleted with the issuance of Amendment No. . However, to maintain numerical continuity between the succeeding sections and existing station procedural references to those Technical Specification sections, 3/4.1.4.2 has been intentionally left blank.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable:

    1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

    2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

    Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

3/4.1 5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system consists of two redundant subsystems and shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one system subsystem inoperable, restore the subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With both system subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that;

   1. The temperature of the sodium pentaborate solution in the storage tank is greater than or equal to 70°F.

   2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-1.

   3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping to be greater than or equal to 70°F.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the continuity of the explosive charge.

2. Determining that the available weight of sodium pentaborate is greater than or equal to 5,776 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

3. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to the IST Program, the minimum flow requirement of 41.2 gpm, per pump, at a pressure of greater than or equal to 1255 psig is met.

d. In accordance with the Surveillance Frequency Control Program by:

1. Initiating one of the standby liquid control system subsystem, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel and verifying that the relief valve does not actuate. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection subsystems shall be tested in accordance with the Surveillance Frequency Control Program.

2. **Demonstrating that all heat traced piping between the storage tank and the injection pumps is unblocked and then draining and flushing the piping with demineralized water.

3. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise of the sodium pentaborate solution in the storage tank after the heaters are energized.

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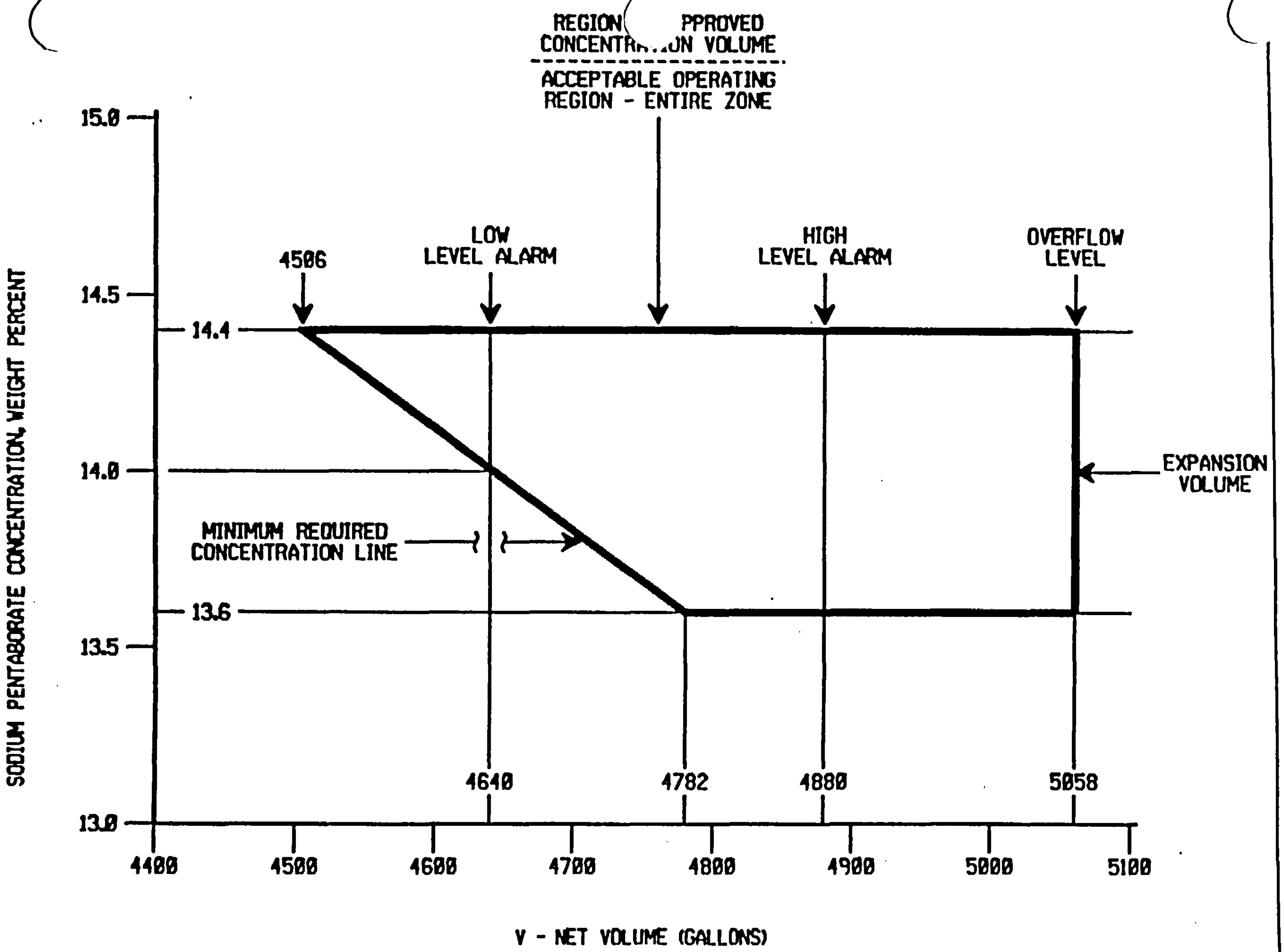
* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F.

** This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.



SODIUM PENTABORATE SOLUTION
VOLUME/CONCENTRATION REQUIREMENTS

FIGURE 3.1.5-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall be less than or equal to the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 24% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT:

a. Once within 12 hours after THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program thereafter.

b. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

3/4.2.2 DELETED

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER.

ACTION:

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the EOC-RPT inoperable limit specified in the CORE OPERATING LIMITS REPORT.

b. With MCPR less than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 24% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT:

a. Once within 12 hours after THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program thereafter.

b. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

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POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 24% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit specified in the CORE OPERATING LIMITS REPORT:

a. Once within 12 hours after THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program thereafter.

b. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

3/4 4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. For the Reactor Vessel Steam Dome Pressure - High Functional Unit and the Reactor Vessel Water Level - Low, Level 3 Functional Unit, the sensor is eliminated from response time testing for RPS circuits.

4.3.1.4 The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 or 3 from OPERATIONAL CONDITION 1 for the Inter-mediate Range Monitors.

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* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

** If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors(b):			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	5(c)	3(d)	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3(d)	3
2. Average Power Range Monitor(e):			
a. Neutron Flux - Upscale, Setdown	2	2	1
	3, 4	2	2
	5(c)	2(d)	3
b. Flow Biased Simulated Thermal Power - Upscale	1	2	4
c. Fixed Neutron Flux - Upscale	1	2	4
d. Inoperative	1, 2	2	1
	3, 4	2	2
	5(c)	2(d)	3
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	4	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. This item intentionally blank			
7. Drywell Pressure - High	1, 2(h)	2	1
8. Scram Discharge Volume Water Level - High			
a. Float Switch	1, 2 5(i)	2 2	1 3
b. Level Transmitter/Trip Unit	1, 2 5(i)	2 2	1 3
9. Turbine Stop Valve - Closure	1(j)	4(k)	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1(j)	2(k)	6
11. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	2 2 2	1 7 3
12. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - This ACTION is deleted

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within one hour.

*Except replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn*.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per the Trip System are 4 APRMS, 6 IRMS and 2 SRMS.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER less than 24% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

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* Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (m)	CHANNEL FUNCTIONAL TEST (m)	CHANNEL CALIBRATION (a) (m)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	(b)			2 3, 4, 5
b. Inoperative	NA		NA	2, 3, 4, 5
2. Average Power Range Monitor(f):				
a. Neutron Flux – Upscale, Setdown	(b)	(l)		2 3, 4, 5
b. Flow Biased Simulated Thermal Power-Upscale	(g)		(d) (e) (h)	1
c. Fixed Neutron Flux - Upscale			(d)	1
d. Inoperative	NA		NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High		(k)		1, 2
4. Reactor Vessel Water Level - Low, Level 3		(k)		1, 2
5. Main Steam Line Isolation Valve - Closure	NA			1
6. This item intentionally blank				
7. Drywell Pressure - High		(k)		1, 2

TABLE 4.3.1.1-1 (Continued)
REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK [(m)]	CHANNEL FUNCTIONAL TEST [(m)]	CHANNEL CALIBRATION [(m)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High:				
a. Float Switch	NA			1, 2, 5[(j)]
b. Level Transmitter/Trip Unit		(k)		1, 2, 5[(j)]
9. Turbine Stop Valve - Closure	NA			1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA			1
11. Reactor Mode Switch Shutdown Position	NA		NA	1, 2, 3, 4, 5
12. Manual Scram	NA		NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.

(c) DELETED

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 24% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated in accordance with the Surveillance Frequency Control Program.

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing recirculation loop flow (APRM % flow).

(h) This calibration shall consist of verifying the 6 ± 0.6 second simulated thermal power time constant.

(i) This item intentionally blank

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(k) Verify the tripset point of the trip unit in accordance with the Surveillance Frequency Control Program.

(l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.

(m) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for one trip system, either

   1) place the inoperable channel(s) in the tripped condition within

      a) 1 hour for trip functions without an OPERABLE channel,
      b) 12 hours for trip functions common to RPS instrumentation, and
      c) 24 hours for trip functions not common to RPS instrumentation,

   or

   2) take the ACTION required by Table 3.3.2-1.

c. With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip system requirement for both trip systems,

   1) place the inoperable channel(s) in one trip system in the tripped condition within one hour, and

   2) a) place the inoperable channel(s) in the remaining trip system in the tripped condition within

         1) 1 hour for trip functions without an OPERABLE channel,
         2) 12 hours for trip functions common to RPS instrumentation, and
         3) 24 hours for trip functions not common to RPS instrumentation,

      or

      b) take the ACTION required by Table 3.3.2-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Radiation detectors are exempt from response time testing. The sensor is eliminated from response time testing for MSIV isolation logic circuits of the following trip functions: Reactor Vessel Water Level - Low Low Low, Level 1; Main Steam Line Pressure - Low; Main Steam Line Flow - High.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low Low, level 2	1, 2, 8, 9, 12, 13, 14, 15, 17, 18	2	1, 2, 3	20
2) Low low Low, Level 1	10, 11, 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	1, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18	2[j]	1, 2, 3	20
c. Reactor Building Exhaust Radiation - High	1, 8, 9, 12 13, 14, 15, 17, 18	3	1, 2, 3	28
d. Manual Initiation	1, 8, 9, 10 11, 12, 13, 14, 15, 16, 17, 18	1	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	19[c]	2	1, 2, 3 and *	26
b. Drywell Pressure - High	19[c]	2[j]	1, 2, 3	26
c. Refueling Floor Exhaust Radiation - High	19[c]	3	1, 2, 3 and *	29
d. Reactor Building Exhaust Radiation - High	19[c]	3	1, 2, 3 and *	28
e. Manual Initiation	19[c]	1	1, 2, 3 and *	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	1	2	1, 2, 3	21
b. Main Steam Line Radiation - High, High	2[(b)]	2	1##, 2##, 3	28
c. Main Steam Line Pressure - Low	1	2	1	22
d. Main Steam Line Flow - High	1	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	1	2	1, 2**, 3**	21
f. Main Steam Line Tunnel Temperature - High	1	2/line	1, 2, 3	21
g. Manual Initiation	1, 2, 17	2	1, 2, 3	25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	7	1/Valve[(e)]	1, 2, 3	23
b. RWCU Δ Flow - High, Timer	7	1/Valve[(e)]	1, 2, 3	23
c. RWCU Area Temperature - High	7	6/Valve[(e)]	1, 2, 3	23
d. RWCU Area Ventilation Δ Temperature-High	7	6/Valve[(e)]	1, 2, 3	23
e. SLCS Initiation	7[(f)]	1/Valve[(e)]	1, 2	23
f. Reactor Vessel Water Level - Low Low, Level 2	7	2/Valve[(e)]	1, 2, 3	23
g. Manual Initiation	7	1/Valve[(e)]	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure (Flow) - High	6	1/Valve[(e)]	1, 2, 3	23
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	6	1/Valve[(e)]	1, 2, 3	23
c. RCIC Steam Supply Pressure - Low	6	2/Valve[(e)]	1, 2, 3	23
d. RCIC Turbine Exhaust Diaphragm Pressure - High	6	2/Valve[(e)]	1, 2, 3	23
e. RCIC Pump Room Temperature - High	6	1/Valve[(e)]	1, 2, 3	23
f. RCIC Pump Room Ventilation Ducts Δ Temperature - High	6	1/Valve[(e)]	1, 2, 3	23
g. RCIC Pipe Routing Area Temperature - High	6	1/Valve[(e)]	1, 2, 3	23
h. RCIC Torus Compartment Temperature-High	6	3/Valve[(e)]	1, 2, 3	23
i. Drywell Pressure - High[(g)]	6	2/Valve[(e)]	1, 2, 3	23
j. Manual Initiation	6[(h)]	1/RCIC System	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure (Flow) - High	5	1/Valve[(e)]	1, 2, 3	23
b. HPCI Steam Line Δ Pressure (Flow) - High, Timer	5	1/Valve[(e)]	1, 2, 3	23
c. HPCI Steam Supply Pressure-Low	5	2/Valve[(e)]	1, 2, 3	23
d. HPCI Turbine Exhaust Diaphragm Pressure - High	5	2/Valve[(e)]	1, 2, 3	23
e. HPCI Pump Room Temperature - High	5	1/Valve[(e)]	1, 2, 3	23
f. HPCI Pump Room Ventilation Ducts Δ Temperature - High	5	1/Valve[(e)]	1, 2, 3	23
g. HPCI Pipe Routing Area Temperature - High	5	1/Valve[(e)]	1, 2, 3	23
h. HPCI Torus Compartment Temperature-High	5	3/Valve[(e)]	1, 2, 3	23
i. Drywell Pressure - High[(g)]	5	2/Valve[(e)]	1, 2, 3	23
j. Manual Initiation	5[(i)]	1/HPCI system	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	3[(j)]	2/Valve[(e)]	1, 2, 3	27
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	3[(j)]	2/Valve[(e)]	1, 2, 3	27
c. Manual Initiation	3	1/Valve[(e)]	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours, and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 26 - Establish SECONDARY CONTAINMENT INTEGRITY with the Filtration, Recirculation and Ventilation System (FRVS) operating within one hour. The action of operating FRVS is not required when the Reactor Vessel Water Level - Low Low, Level 2 instrumentation is inoperable as long as the following conditions are met:

a) the reactor water level is maintained at least 22 feet 2 inches over the top of the reactor pressure vessel flange,

b) the suppression pool level is maintained at greater than or equal to 5 inches indicated level,

c) at least one channel of the suppression pool high level alarm is operable, and

d) the spent fuel pool gates are removed.

ACTION 27 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

ACTION 28 - Place the inoperable channel in the tripped condition or close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 29 - Place the inoperable channel in the tripped condition or establish SECONDARY CONTAINMENT INTEGRITY with the Filtration, Recirculation, and Ventilation System (FRVS) operating within one hour.

TABLE 3.3.2-1 (Continued)

NOTES

* When handling recently irradiated fuel in the secondary containment and during operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

Below 20% of RATED THERMAL POWER the Main Steamline Radiation Monitor setpoints shall not exceed the values determined using normal full power background radiation levels with the hydrogen water chemistry (HWC) system shut down. After reaching 20% of RATED THERMAL POWER the normal full power background radiation level and associated trip setpoints may be increased to levels previously measured during full power operation with hydrogen injection. Prior to decreasing below 20% of RATED THERMAL POWER the background level and associated setpoint shall be returned to the normal full power values. If the Main Steamline Radiation Monitor setpoints have been increased for HWC operation and a power reduction event occurs so that the reactor power is below 20% of RATED THERMAL POWER without the required setpoint change, control rod motion shall be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made.

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also trips and isolates the mechanical vacuum pumps.

(c) Also starts the Filtration, Recirculation and Ventilation System (FRVS).

(d) DELETED

(e) Sensors arranged per valve group, not per trip system.

(f) Closes only RWCU system isolation valve(s) HV-F001 and HV-F004.

(g) Requires system steam supply pressure-low coincident with drywell pressure-high to close turbine exhaust vacuum breaker valves.

(h) Manual isolation closes HV-F008 only, and only following manual or automatic initiation of the RCIC system.

(i) Manual isolation closes HV-F003 and HV-F042 only, and only following manual or automatic initiation of the HPCI system.

(j) Trip functions common to RPS instrumentation.

Pages 3/4 3-18 through 3/4 3-21 have been intentionally omitted

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
2) Low Low Low, Level 1	≥ -129.0 inches*	≥ -136.0 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Building Exhaust Radiation - High	≤ $1x10^{-3}$µCi/cc	≤ $1.2x10^{-3}$µCi/cc
d. Manual Initiation	NA	NA
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Refueling Floor Exhaust Radiation - High	≤ $2x10^{-3}$µCi/cc	≤ $2.4x10^{-3}$µCi/cc
d. Reactor Building Exhaust Radiation - High	≤ $1x10^{-3}$µCi/cc	≤ $1.2x10^{-3}$µCi/cc
e. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129.0 inches*	≥ -136.0 inches
b. Main Steam Line Radiation - High, High###	≤ 3.0 X full power background	≤ 3.6 X full power background
c. Main Steam Line Pressure - Low	≥ 756.0 psig	≥ 736.0 psig
d. Main Steam Line Flow - High	≤ 162.8 psid	≤ 169.3 psid

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
MAIN STEAM LINE ISOLATION (Continued)		
e. Condenser Vacuum - Low	$\geq$ 8.5 inches Hg vacuum	$\geq$ 7.6 inches Hg vacuum
f. Main Steam Line Tunnel Temperature - High	$\leq$ 160°F	$\leq$ 172°F
g. Manual Initiation	NA	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. RWCU Δ Flow - High	$\leq$ 56.3 gpm	$\leq$ 61.3 gpm
b. RWCU Δ Flow - High, Timer	45.0 seconds $\leq t \leq$ 47.0 seconds	45.0 seconds $\leq t \leq$ 47.0 seconds
c. RWCU Area Temperature - High	$\leq$ 160°F, 140°F or 135°F***	$\leq$ 172°F, 152°F or 147°F***
d. RWCU/Area Ventilation Δ Temperature - High	$\leq$ 60°F	$\leq$ 70°F
e. SLCS Initiation	NA	NA
f. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -38.0 inches*	$\geq$ -45.0 inches
g. Manual Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Δ Pressure (Flow) - High	$\leq$ 598" H_2O	$\leq$ 611" H_2O
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	3.0 seconds $\leq t \leq$ 13.0 seconds	3.0 seconds $\leq t \leq$ 13.0 seconds
c. RCIC Steam Supply Pressure - Low	$\geq$ 64.5 psig	$\geq$ 56.5 psig
d. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10.0 psig	$\leq$ 20.0 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)		
e. RCIC Pump Room Temperature - High	$\leq$ 160°F	$\leq$ 172°F
f. RCIC Pump Room Ventilation Duct Δ Temperature - High	$\leq$ 70°F	$\leq$ 80°F
g. RCIC Pipe Routing Area Temperature - High	$\leq$ 160°F#	$\leq$ 172°F#
h. RCIC Torus Compartment Temperature - High	$\leq$ 128°F#	$\leq$ 140°F#
i. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
j. Manual Initiation	NA	NA
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Δ Pressure (Flow) - High	$\leq$ 1032 inches H_2O	$\leq$ 1064 inches H_2O
b. HPCI Steam Line Δ Pressure (Flow) - High, Timer	3.0 seconds $\leq$ t $\leq$ 13.0 seconds	3.0 seconds $\leq$ t $\leq$ 13.0 seconds
c. HPCI Steam Supply Pressure - Low	$\geq$ 100.0 psig	$\geq$ 90.0 psig
d. HPCI Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10.0 psig	$\leq$ 20.0 psig
e. HPCI Pump Room Temperature - High	$\leq$ 160°F	$\leq$ 172°F
f. HPCI Pump Room Ventilation Ducts Δ Temperature - High	$\leq$ 70°F	$\leq$ 80°F
g. HPCI Pipe Routing Area Temperature - High	$\leq$ 160°F##	$\leq$ 172°F##
h. HPCI Torus Compartment Temperature - High	$\leq$ 128°F##	$\leq$ 140°F##
i. Drywell Pressure High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
j. Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	$\geq$ 12.5 inches*	$\geq$ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	$\leq$ 82.0 psig	$\leq$ 102.0 psig
c. Manual Initiation	NA	NA

* See Bases Figure B 3/4 3-1.

*** These setpoints are as follows:

160°F - RWCU pipe chase room 4402
140°F - RWCU pump room and heat exchanger rooms
135°F - RWCU pipe chase room 4505

\# 30 minute time delay.

\#\# 15 minute time delay.

\#\#\# Below 20% of RATED THERMAL POWER the Main Steamline Radiation Monitor setpoints shall not exceed the values determined using normal full power background radiation levels with the hydrogen water chemistry (HWC) system shut down. After reaching 20% of RATED THERMAL POWER the normal full power background radiation level and associated trip setpoints may be increased to levels previously measured during full power operation with hydrogen injection. Prior to decreasing below 20% of RATED THERMAL POWER the background level and associated setpoint shall be returned to the normal full power values. If the Main Steamline Radiation Monitor setpoints have been increased for HWC operation and a power reduction event occurs so that the reactor power is below 20% of RATED THERMAL POWER without the required setpoint change, control rod motion shall be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made.

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TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK [(c)]	CHANNEL FUNCTIONAL TEST [(c)]	CHANNEL CALIBRATION [(c)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1) Low Low, Level 2				1, 2, 3
2) Low Low Low, Level 1				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. Reactor Building Exhaust Radiation - High				1, 2, 3
d. Manual Initiation	NA	(a)	NA	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2				1, 2, 3 and *
b. Drywell Pressure - High				1, 2, 3
c. Refueling Floor Exhaust Radiation - High				1, 2, 3 and *
d. Reactor Building Exhaust Radiation - High				1, 2, 3 and *
e. Manual Initiation	NA	(a)	NA	1, 2, 3 and *
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level – Low Low Low, Level 1				1, 2, 3
b. Main Steam Line Radiation - High, High				1, 2, 3
c. Main Steam Line Pressure - Low				1
d. Main Steam Line Flow - High				1, 2, 3
e. Condenser Vacuum - Low				1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	NA			1, 2, 3
g. Manual Initiation	NA	(a)	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION		CHANNEL CHECK [c]	CHANNEL FUNCTIONAL TEST [c]	CHANNEL CALIBRATION [c]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4.	REACTOR WATER CLEANUP SYSTEM ISOLATION				
a.	RWCU Δ Flow - High				1, 2, 3
b.	RWCU Δ Flow – High, Timer	NA			1, 2, 3
c.	RWCU Area Temperature - High	NA			1, 2, 3
d.	RWCU Area Ventilation Δ Temperature - High	NA			1, 2, 3
e.	SLCS Initiation	NA	(b)	NA	1, 2
f.	Reactor Vessel Water Level - Low Low, Level 2				1, 2, 3
g.	Manual Initiation	NA	(a)	NA	1, 2, 3
5.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a.	RCIC Steam Line Δ Pressure (Flow) - High	NA			1, 2, 3
b.	RCIC Steam Line Δ Pressure (Flow) – High, Timer	NA			1, 2, 3
c.	RCIC Steam Supply Pressure - Low	NA			1, 2, 3
d.	RCIC Turbine Exhaust Diaphragm Pressure - High	NA			1, 2, 3
e.	RCIC Pump Room Temperature - High	NA			1, 2, 3
f	RCIC Pump Room Ventilation Ducts Δ Temperature - High	NA			1, 2, 3
g.	RCIC Pipe Routing Area Temperature - High	NA			1, 2, 3
h.	RCIC Torus Compartment Temperature -High	NA			1, 2, 3
i.	Drywell Pressure - High				1, 2, 3
j.	Manual Initiation	NA		NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (c)	CHANNEL FUNCTIONAL TEST (c)	CHANNEL CALIBRATION (c)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure (Flow) - High	NA			1, 2, 3
b. HPCI Steam Line Δ Pressure (Flow) – High, Timer	NA			1, 2, 3
c. HPCI Steam Supply Pressure - Low	NA			1, 2, 3
d. HPCI Turbine Exhaust Diaphragm Pressure - High	NA			1, 2, 3
e. HPCI Pump Room Temperature - High	NA			1, 2, 3
f HPCI Pump Room Ventilation Ducts Δ Temperature - High	NA			1, 2, 3
g. HPCI Pipe Routing Area Temperature - High	NA			1, 2, 3
h. HPCI Torus Compartment Temperature -High	NA			1, 2, 3
i. Drywell Pressure - High	NA			1, 2, 3
j. Manual Initiation	NA		NA	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3				1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA			1, 2, 3
c. Manual Initiation	NA	(a)	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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* When handling recently irradiated fuel in the secondary containment and during operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

(a) Manual initiation switches shall be tested in accordance with the Surveillance Frequency Control Program. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested in accordance with the Surveillance Frequency Control Program.

(c) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. ECCS actuation instrumentation is eliminated from response time testing.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)(e)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)(e)	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	4/division(f)	1, 2, 3	31
		4*, 5*	32
d. Core Spray Pump Discharge Flow - Low (Bypass)	1/subsystem	1, 2, 3, 4*, 5*	37
e. Core Spray Pump Start Time Delay - Normal Power	1/subsystem	1, 2, 3, 4*, 5*	31
f. Core Spray Pump Start Time Delay - Emergency Power	1/subsystem	1, 2, 3, 4*, 5*	31
g. Manual Initiation	1/division(b)(g)	1, 2, 3, 4*, 5*	33
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2/valve	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2/valve	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	1/valve	1, 2, 3	31
		4*, 5*	32
d. LPCI Pump Discharge Flow - Low (Bypass)	1/pump	1, 2, 3, 4*, 5*	37
e. LPCI Pump Start Time Delay - Normal Power	1/pump(i)	1, 2, 3, 4*, 5*	31
f. Manual Initiation	1/subsystem	1, 2, 3, 4*, 5*	33
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#			
a. Reactor Vessel Water Level - Low Low Level 2	4	1, 2, 3	34
b. Drywell Pressure - High	4	1, 2, 3	34
c. Condensate Storage Tank Level - Low	2(c)	1, 2, 3	35
d. Suppression Pool Water Level - High	2(c)	1, 2, 3	35
e. Reactor Vessel Water Level - High, Level 8	4(d)	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	1	1, 2, 3	37
g. Manual Initiation	1/system	1, 2, 3	33
4. AUTOMATIC DEPRESSURIZATION SYSTEM##			
a. Reactor Vessel Water Level - Low Low Low, Level 1	4	1, 2, 3	30
b. Drywell Pressure - High	4	1, 2, 3	30
c. ADS Timer	2	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	1/pump	1, 2, 3	31

TABLE 3.3.3-1 (Cont'd)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM##			
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	31
f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	2	1, 2, 3	31
g. ADS Drywell Pressure Bypass Timer	4	1, 2, 3	31
h. ADS Manual Inhibit Switch	2	1, 2, 3	31
i. Manual Initiation	4	1, 2, 3	33

	TOTAL NO. OF CHANNELS(h)	CHANNELS TO TRIP(h)	MINIMUM CHANNELS OPERABLE(h)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
1. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	4/bus	2/bus	3/bus	1, 2, 3, 4**, 5**	36
2. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	2/source/bus	2/source/bus	2/source/bus	1, 2, 3, 4**, 5**	36

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.
(b) Also actuates the associated emergency diesel generators.
(c) One trip system. Provides signal to HPCI pump suction valve only.
(d) Provides a signal to trip HPCI pump turbine only.
(e) In divisions 1 and 2, the two sensors are associated with each pump and valve combination. In divisions 3 and 4, the two sensors are associated with each pump only.
(f) Division 1 and 2 only.
(g) In divisions 1 and 2, manual initiation is associated with each pump and valve combination; in divisions 3 and 4, manual initiation is associated with each pump only.
(h) Each voltage detector is a channel.
(i) Start time delay is applicable to LPCI Pump C and D only.
* When the system is required to be OPERABLE per Specification 3.5.2.
** Required when ESF equipment is required to be OPERABLE.
# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.
Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the associated system inoperable.

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ECCS inoperable within 24 hours.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 24 hours.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the associated ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the HPCI system inoperable.

b. With more than one channel inoperable, declare the HPCI system inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 24 hours or declare the HPCI system inoperable.

ACTION 36 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACITON 37 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function requirement, open the minimum flow bypass valve within one hour. Restore the inoperable channel to OPERABLE status within 7 days or declare the associated ECCS inoperable.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Pressure - Low	461 psig	≤ 481 psig and ≥ 441 psig
d. Core Spray Pump Discharge Flow - Low (Bypass)	≥ 775 gpm	≥ 650 gpm
e. Core Spray Pump Start Time Delay - Normal Power	10 seconds	≥ 9 seconds and ≤ 11 seconds
f. Core Spray Pump Start Time Delay - Emergency Power	6 seconds	≥ 5 Seconds and ≤ 7 Seconds
g. Manual Initiation	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Pressure - Low (Permissive)	450 psig	≤ 460 psig and ≥ 440 psig
d. LPCI Pump Discharge Flow - Low (Bypass)	≥ 1250 gpm	≥ 1100 gpm
e. LPCI Pump Start Time Delay - Normal Power	5 seconds	≥ 4 seconds and ≤ 6 seconds
f. Manual Initiation	NA	NA
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - (Low Low, Level 2)	≥ -38 inches*	≥ -45 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Condensate Storage Tank Level - Low	≥ 67,675 gallons	≥ 64,291 gallons
d. Suppression Pool Water Level - High	≤ 78.5 inches	≤ 80.3 inches
e. Reactor Vessel Water Level - High, Level 8	≤ 54 inches	≤ 61 inches
f. HPCI Pump Discharge Flow - Low (Bypass)	≥ 550 gpm	≥ 500 gpm
g. Manual Initiation	NA	NA

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
4.	AUTOMATIC DEPRESSURIZATION SYSTEM		
a.	Reactor Water Level - Low Low Low, Level 1	$\geq$-129 inches*	$\geq$-136 inches
b.	Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c.	ADS Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d.	Core Spray Pump Discharge Pressure - High	145 psig	$\leq$ 155 psig $\geq$ 125 psig
e.	RHR LPCI Mode Pump Discharge Pressure-High	125 psig	$\leq$ 135 psig $\geq$ 115 psig
f.	Reactor Vessel Water Level-Low, Level 3	$\geq$ 12.5 inches	$\geq$ 11.0 inches
g.	ADS Drywell Pressure Bypass Timer	$\leq$ 5.0 minutes	$\leq$ 5.5 minutes
h.	ADS Manual Inhibit Switch	NA	NA
i.	Manual Initiation	NA	NA
5.	LOSS OF POWER		
a.	4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	a. 4.16 kv Basis - 2975 ± 30 volts	2975 ± 63 volts
		b. 120 v Basis - 85 ± 0.85 volts	85 ± 1.8 volts
b.	4.16 kv Emergency Bus Undervoltage (Degraded Voltage)**	a. 4.16 kv Basis - $\geq$ 3857 volts	$\geq$ 3857 volts
		b. 120 v Basis - $\geq$ 110.2 volts	$\geq$ 109.0 volts
		c. 20 sec @ 109.0 volts	20 + 15, - 5 sec @ 109.0 volts

* See Bases Figure B 3/4 3-1.

** This is a solid state voltage relay. The voltages shown are the minimum that will not result in a trip. Some voltage conditions will result in decreased trip times.

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TABLE 4.3.3.1-1
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK [(a)]	CHANNEL FUNCTIONAL TEST [(a)]	CHANNEL CALIBRATION [(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level – Low Low Low, Level 1				1, 2, 3, 4*, 5*
b. Drywell Pressure - High				1, 2, 3
c. Reactor Vessel Pressure - Low				1, 2, 3, 4*, 5*
d. Core Spray Pump Discharge Flow - Low (Bypass)				1, 2, 3, 4*, 5*
e. Core Spray Pump Start Time Delay - Normal Power	NA			1, 2, 3, 4*, 5*
f. Core Spray Pump Start Time Delay - Emergency Power	NA			1, 2, 3, 4*, 5*
g. Manual Initiation	NA		NA	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level – Low Low Low, Level 1				1, 2, 3, 4*, 5*
b. Drywell Pressure - High				1, 2, 3
c. Reactor Vessel Pressure – Low (Permissive)				1, 2, 3, 4*, 5*
d. LPCI Pump Discharge Flow - Low (Bypass)				1, 2, 3, 4*, 5*
e. LPCI Pump Start Time Delay - Normal Power	NA			1, 2, 3, 4*, 5*
f. Manual Initiation	NA		NA	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#				
a. Reactor Vessel Water Level – Low Low, Level 2				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. Condensate Storage Tank Level - Low				1, 2, 3
d. Suppression Pool Water Level - High				1, 2, 3
e. Reactor Vessel Water Level - High, Level 8				1, 2, 3
f. HPCI Pump Discharge Flow – Low (Bypass)				1, 2, 3
g. Manual Initiation	NA		NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK [(a)]	CHANNEL FUNCTIONAL TEST [(a)]	CHANNEL CALIBRATION [(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. AUTOMATIC DEPRESSURIZATION SYSTEM##				
a. Reactor Vessel Water Level – Low Low Low, Level 1				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. ADS Timer	NA			1, 2, 3
d. Core Spray Pump Discharge Pressure - High				1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High				1, 2, 3
f. Reactor Vessel Water Level - Low, Level 3				1, 2, 3
g. ADS Drywell Pressure Bypass Timer	NA			1, 2, 3
h. ADS Manual Inhibit Switch	NA		NA	1, 2, 3
i. Manual initiation	NA		NA	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	NA	NA		1, 2, 3, 4**, 5**
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)				1, 2, 3, 4**, 5**

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.
* When the system is required to be OPERABLE per Specification 3.5.2.
** Required OPERABLE when ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.
\## Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, and:

   1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within one hour, or if this action will initiate a pump trip, declare the trip system inoperable.

   2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1. Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies in accordance with the Surveillance Frequency Control Program.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure - High	2

(a) One channel may be placed in an inoperable status for up to 2 hours for required surveillance provided the other channel is OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	$\geq$ -38 inches*	$\geq$ -45 inches
2. Reactor Vessel Pressure - High	$\leq$ 1071 psig	$\leq$ 1086 psig

*See Bases Figure B3/4 3-1.

TABLE 4.3.4.1-1

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END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 12 hours.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 hours.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies in accordance with the Surveillance Frequency Control Program.

4.3.4.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested in accordance with the Surveillance Frequency Control Program.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured in accordance with the Surveillance Frequency Control Program.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Turbine Stop Valve - Closure	2[b]
2. Turbine Control Valve-Fast Closure	2[b]

[a] A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

[b] This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER less than 24% of RATED THERMAL POWER.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure	$\leq$ 5% closed	$\leq$ 7% closed
2. Turbine Control Valve-Fast Closure	$\geq$ 530 psig	$\geq$ 465 psig

TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milleseconds)
1. Turbine Stop Valve-Closure	$\leq$ 175
2. Turbine Control Valve-Fast Closure	$\leq$ 175

TABLE 4.3.4.2.1-1

INFORMATION ON THIS PAGE HAS BEEN DELETED

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	4(b)	50
b. Reactor Vessel Water Level - High, Level 8	4(b)	50
c. Condensate Storage Tank Water Level - Low(e)	2(c)	51
d. Manual Initiation	1(d)	52

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided all other channels monitoring that parameter are OPERABLE.
(b) One trip system with one-out-of-two twice logic.
(c) One trip system with one-out-of-two logic.
(d) One trip system with one channel.
(e) Initiates RCIC suction switchover from the condensate storage tank to the torus.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the RCIC system inoperable.

b. With more than one channel inoperable, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 24 hours or declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the RCIC system inoperable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
b. Reactor Vessel Water Level - High, Level 8	≤ 54 inches*	≤ 61 inches
c. Condensate Storage Tank Level - Low	≥ 67,675 gallons	≥ 64,291 gallons
d. Manual Initiation	NA	NA

*See Bases Figure B 3/4 3-1.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK [b]	CHANNEL FUNCTIONAL TEST [b]	CHANNEL CALIBRATION [b]
a. Reactor Vessel Water Level – Low Low, Level 2			
b. Reactor Vessel Water Level - High, Level 8			
c. Condensate Storage Tank Level - Low	NA		
d. Manual Initiation	NA	[a]	NA

(a) Manual initiation switches shall be tested in accordance with the Surveillance Frequency Control Program. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program as part of circuitry required to be tested for automatic system actuation.

(b) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1. The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 for the Source Range Monitors or the Intermediate Range Monitors.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[a]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in[b]	3	2	61
	2	5	61
b. Upscale[c]	3	2	61
	2	5	61
c. Inoperative[c]	3	2	61
	2	5	61
d. Downscale[d]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[e]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High (Float Switch)	2	1, 2, 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. REACTOR MODE SWITCH SHUTDOWN POSITION	2	3, 4	63

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale		
i. Flow Biased	≤ 0.66 (w-Δw) + 65%*	≤ 0.66 (w-Δw) + 68%*
ii. High Flow Clamped	≤ 116%	≤ 119%
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.57(w-Δw) + 53%*	≤ 0.57(w-Δw) + 56%*
b. Inoperative	NA	NA
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 11% of RATED THERMAL POWER	≤ 13% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High (Float Switch)	109'1" (North Volume) 108'11.5" (South Volume)	109'3" (North Volume) 109'1.5" (South Volume)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 111% of rated flow	≤ 114% of rated flow
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

* The rod block function is varied as a function of recirculation loop flow (w) and Δw which is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow.

TABLE 4.3.6-1
CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (f)	CHANNEL FUNCTIONAL TEST (f)	CHANNEL CALIBRATION(a) (f)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	(c)		1*
b. Inoperative	NA	(c)	NA	1*
c. Downscale	NA	(c)		1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	NA			1
b. Inoperative	NA		NA	1, 2, 5
c. Downscale	NA			1
d. Neutron Flux - Upscale, Startup	NA			2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA		NA	2, 5
b. Upscale	NA			2, 5
c. Inoperative	NA		NA	2, 5
d. Downscale	NA			2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA		NA	2, 5
b. Upscale	NA			2, 5
c. Inoperative	NA		NA	2, 5
d. Downscale	NA			2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High (Float Switch)	NA			1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA			1
b. Inoperative	NA		NA	1
c. Comparator	NA			1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	(e)	NA	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. DELETED

c. Includes reactor manual control multiplexing system input.

d. DELETED

e. Not required to be performed until 1 hour after reactor mode switch is in the shutdown position.

f. Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

* With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

INSTRUMENTATION

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies in accordance with the Surveillance Frequency Control Program.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Control Room Ventilation Radiation Monitor	2/intake	1,2,3 and *	$\leq 2\times10^{-5}$ µC/cc**	71
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	1	#	$\geq$ 5 mR/hr and $\leq$ 20 mR/hr[(a)]	72
2) Spent Fuel Storage Pool	1	##	$\geq$ 5 mR/hr and $\leq$ 20 mR/hr[(a)]	72
b. Control Room Direct Radiation Monitor	1	At all times	2.5 mR/hr[(a)]	72
3. Reactor Auxiliaries Cooling Radiation Monitor	1	At all times	9×10^{-5} µC/cc[(a)]	73
4. Safety Auxiliaries Cooling Radiation Monitor	1/loop	At all times	6×10^{-5} µC/cc[(a)]	73
5. Offgas Pre-treatment Radiation Monitor	1	***	(b)	74

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATION

*When recently irradiated fuel is being handled in the secondary containment and during operations with the potential for draining the reactor vessel.

**Activates control room emergency filtration system.

***When the offgas treatment system is operating.

#With fuel in the new fuel storage vault.

##With fuel in the spent fuel storage pool.

(a)Alarm only.

(b)Alarm setpoint to be set in accordance with Specification 3.11.2.7.

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 71 -

a. With one of the required monitors inoperable, place the inoperable channel in the tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within one hour.

ACTION 72 - With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, obtain and analyze at least one sample of the monitored parameter at least once per 24 hours.

ACTION 74 - With the number of channels OPERABLE less than required by Minimum Channels OPERABLE requirement, release(s) via this pathway may continue for up to 30 days provided:

a. The offgas system is not bypassed, and

b. Grab samples are taken at least once per 8 hours and analyzed within the following 4 hours;

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

TABLE 4.3.7.1-1
RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK [(a)]	CHANNEL FUNCTIONAL TEST [(a)]	CHANNEL CALIBRATION [(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Control Room Ventilation Radiation Monitor				1, 2, 3, and *
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault				#
2) Spent Fuel Storage Pool				##
b. Control Room Direct Radiation Monitor				At all times
3. Reactor Auxiliaries Cooling Radiation Monitor				At all times
4. Safety Auxiliaries Cooling Radiation Monitor				At all times
5. Offgas Pre-treatment Radiation Monitor				**

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

#With fuel in the new fuel storage vault.

##With fuel in the spent fuel storage pool.

*When recently irradiated fuel is being handled in the secondary containment and during operations with the potential for draining the reactor vessel.

**When the offgas treatment system is operating.

INSTRUMENTATION

3.3.7.2 DELETED

3.3.7.3 DELETED

PAGES 3/4 3-69 THRU 3/4 3-73 ARE DELETED

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system instrumentation and controls shown in Table 3.3.7.4-1 and Table 3.3.7.4-2 shall be OPERABLE.

APPLICABILITY OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown system controls less than required in Table 3.3.7.4-2, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted by Table 4.3.7.4-1.

4.3.7.4.2 At least one of each of the above remote shutdown control switch(es) and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) in accordance with the Surveillance Frequency Control Program.

TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE*
1. Reactor Vessel Pressure	2
2. Reactor Vessel Water Level	2
3. Safety/Relief Valve Position, (3) valves	1/valve
4. Suppression Chamber Water Level	2
5. Suppression Chamber Water Temperature	2
6. RHR System Flow	1
7. Safety Auxiliaries Cooling System Flow	1
8. Safety Auxiliaries Cooling System Temperature	1
9. RCIC System Flow	1
10. RCIC Turbine Speed	1
11. RCIC Turbine Bearing Oil Pressure Low Indication	1
12. RCIC High Pressure/Low Pressure Turbine Bearing Temperature High Indication	1
13. Condensate Storage Tank Level Low-Low Indication	1
14. Standby Diesel Generator 1AG400 Breaker Indication	1

*Either primary location (Remote Shutdown Panel, 10C399) or alternate location.

TABLE 3.3.7.4-1 (Continued)

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT (Continued)	MINIMUM INSTRUMENTS OPERABLE*
15. Standby Diesel Generator 1BG400 Breaker Indication	1
16. Standby Diesel Generator 1CG400 Breaker Indication	1
17. Standby Diesel Generator 1DG400 Breaker Indication	1
18. Switchgear Room Cooler 1AVH401 Status Indication	1
19. Switchgear Room Cooler 1BVH401 Status Indication	1
20. Switchgear Room Cooler 1CVH401 Status Indication	1
21. Switchgear Room Cooler 1DVH401 Status Indication	1

*Either primary location (Remote Shutdown Panel 10C399) or alternate location.

TABLE 3.3.7.4-2

REMOTE SHUTDOWN SYSTEMS CONTROLS

CHANNEL TRANSFER SWITCHES - REMOTE SHUTDOWN PANEL (RSP)(1)

1SV-HSS-4410A	Control -	Class 1E Channel A Transfer Switch
1SV-HSS-4410B	Control -	Class 1E Channel B Transfer Switch
1SV-HSS-4410C	Control -	Class 1E Channel C Transfer Switch
1SV-HSS-4410D	Control -	Class 1E Channel D Transfer Switch
1SV-HSS-4410N	Control -	Non-Class 1E Transfer Switch

RCIC SYSTEM - RSP

1FC-HV-4282	Control -	RCIC Turbine Trip/Throttle Valve
1FC-HV-F045	Control -	RCIC Turbine Shutoff Valve
1FC-HV-F008	Control -	RCIC Steam Supply Outboard Isolation Valve
1FC-HV-F007	Control -	RCIC Steam Supply Inboard Isolation Valve
1BD-HV-F031	Control -	Suppression Pool to RCIC Pump Suction Valve
1BD-HV-F010	Control -	Condensate Storage Tank to RCIC Pump Suction Valve
1BD-SV-F019	Control -	RCIC Pump Discharge Minimum Flow Valve
1BD-HV-F046	Control -	RCIC Turbine Cooling Water Supply Valve
1BD-HV-F013	Control -	RCIC Pump Discharge to Feedwater Line Isolation Valve
1FC-HV-F076	Control-	RCIC Steam Line Inboard Isolation Valve
1BD-HV-F012(2)	Indication -	RCIC Pump Discharge Valve
1BD-HV-F022(3)	Indication -	Test Return Valve to Condensate Storage Tank
1FC-HV-F059(2)	Indication -	RCIC Turbine Exhaust to Suppression Pool Valve
1FC-HV-F060(2)	Indication -	RCIC Condenser Vacuum Pump Discharge Valve
1FC-HV-F062(2)	Indication -	RCIC Turbine Exhaust Outboard Vacuum Breaker Isolation Valve
1FC-HV-F084(2)	Indication -	RCIC Turbine Exhaust Inboard Vacuum Breaker Isolation Valve
1FC-HV-F025(3)	Indication -	RCIC Condensate Pot Drain to Main Condenser Valve
1FC-HV-F004(3)	Indication -	RCIC Vacuum Tank Condensate Pump Discharge to Clean Rad Waste Valve
1BD-BP228(4)	Indication -	ECCS (RCIC) Jockey Pump BP228
1FC-OP220	Control -	RCIC Vacuum Tank Condensate Pump OP220
1FC-OP219	Control -	RCIC Gland Seal Condenser Vacuum Pump OP219
1FC-FIC-4158	Control -	RCIC System Injection Flow

RHR SYSTEM - RSP

1BC-HV-F006B	Control -	RHR Pump BP202 Suction From Recirc Line Valve
1BC-HV-F004B	Control -	RHR Pump BP202 Suction From Suppression Pool Valve

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEMS CONTROLS

RHR SYSTEM - RSP (Cont.)

1BC-HV-F007B	Control -	RHR Pump BP202 Minimum Flow Valve to Suppression Pool
1BC-HV-F048B	Control -	RHR Loop B Heat Exchanger Bypass Valve
1BC-HV-F015B	Control -	RHR Loop B Shutdown Cooling Return Valve
1BC-HV-F009	Control -	RHR Shutdown Cooling Suction From Recirc Line Inboard Isolation Valve
1BC-HV-F008	Control -	RHR Shutdown Cooling Suction From Recirc Line Outboard Isolation Valve
1BC-HV-F122B	Control -	RHR Loop B Shutdown Cooling Injection Check Valve Bypass Valve
1BC-HV-4439	Control -	RHR Discharge to Liquid Radwaste Reactor Building Isolation Valve
1BC-HV-F024B	Control -	RHR Pump BP202 Test Return Valve to Suppression Pool
1BC-HV-F047B	Control -	RHR Loop B Heat Exchanger Shell Side Inlet Valve
1BC-HV-F003B	Control -	RHR Loop B Heat Exchanger Shell Side Outlet Valve
1BC-HV-F049	Control -	RHR Discharge to Liquid Radwaste Inboard Isolation Valve
1BC-HV-F040	Control -	RHR Discharge to Liquid Radwaste Outboard Isolation Valve
1BC-HV-F006A(3)	Indication-	RHR Pump AP202 Suction From Recirc Line Valve
1BC-HV-F010B(3)	Indication -	RHR Pump DP202 Test Return Valve to Suppression Pool
1BC-HV-F016B(3)	Indication -	RHR Loop B Containment Spray Outboard Isolation Valve
1BC-HV-F027B(3)	Indication -	RHR Loop B Suppression Pool Spray Line Isolation Valve
1BC-HV-F017B(3)	Indication -	RHR Low Pressure Coolant Injection Loop B Injection Valve
1BC-HV-F004D(2)	Indication -	RHR Pump DP202 Suction From Suppression Pool Valve
1BC-HV-F021A(3)	Indication -	RHR Loop A Containment Spray Inboard Isolation Valve
1BC-HV-F021B(3)	Indication -	RHR Loop B Containment Spray Inboard Isolation Valve
1BC-BP202	Control -	RHR Pump BP202
1BC-HSS-4416B	Control -	Transfer Switch For RHR Pump BP202
1BC-DP228(4)	Indication -	ECCS (RHR B) Jockey Pump DP228

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEMS CONTROLS

RHR SYSTEM - REDUNDANT CONTROLS

1BC-HV-F006A	Local Control -	RHR Pump AP202 Suction From Recirc Line Valve
1BC-HV-F004A	Local Control -	RHR Pump AP202 Suction From Suppression Pool Valve
1BC-HV-F048A	Local Control -	RHR Loop A Heat Exchanger Bypass Valve
1BC-HV-F015A	Local Control -	RHR Loop A Shutdown Cooling Return Valve
1BC-HV-F024A	Local Control -	RHR Pump AP202 Test Return Valve to Suppression Pool
1BC-HV-F047A	Local Control -	RHR Loop A Heat Exchanger Shell Side Inlet Valve
1BC-HV-F003A	Local Control -	RHR Loop A Heat Exchanger Shell Side Outlet Valve
1BC-AP202	Local Control -	RHR Pump AP202

SACS - RSP

1EG-HV-2522B/ 1EG-HV-2496B[5]	Control -	SACS Loop B to Turbine Auxiliaries Cooling System (TACS) Inboard Supply and Return Valves
1EG-HV-2522D/ 1EG-HV-2496D[6]	Control -	SACS Loop B to TACS Outboard Supply and Return Valves
1EG-HV-2512B	Control -	RHR Loop B Heat Exchanger Tube Side Outlet Valve
1EG-HV-2491B	Control -	SACS Loop B Heat Exchanger B1E201, Inlet Valve
1EG-HV-2494B	Control -	SACS Loop B Heat Exchanger B2E201, Inlet Valve
1EG-HV-2520B[7][2]	Indication -	RHR Pump BP202 Seal and Motor Bearing Coolers Cooling Water Supply Valve
1EG-BP210	Control -	SACS Loop B Pump BP210
1EG-HSS-2485B	Control -	Transfer Switch For SACS Loop B Pump BP210
1EG-DP210	Control -	SACS Loop B Pump DP210
1EG-HSS-2485D	Control -	Transfer Switch For SACS Loop B Pump DP210

SACS - REDUNDANT CONTROLS

1EG-HV-2496A	Local Control -	SACS Loop A Return From TACS Inboard Valve
1EG-HV-2496C	Local Control -	SACS Loop A Return From TACS Outboard Valve
1EG-HV-2512A	Local Control -	RHR Loop A Heat Exchanger Tube Side Outlet Valve

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEMS CONTROLS

SACS - REDUNDANT CONTROLS (Cont.)

1EG-AP210	Local Control -	SACS Loop A Pump AP210
1EG-CP210	Local Control -	SACS Loop A Pump CP210

STATION SERVICE WATER SYSTEM (SSWS) - RSP

1EA-HV-2204	Control -	Reactor Auxiliaries Cooling System (RACS) Heat Exchanger Supply Valve (From SACS Loop B)
1EA-HV-2355B	Control -	SACS Loop B Heat Exchanger B2E201 Outlet Valve
1EA-HV-2371B	Control -	SACS Loop B Heat Exchanger B1E201 Outlet Valve
1EA-HV-2357B	Control -	SACS Loop B to Cooling Tower Valve
1EA-HV-2198B	Control -	SSWS Pump BP502 Discharge Valve
1EA-HV-2198D	Control -	SSWS Pump DP502 Discharge Valve
1EA-HV-2197B	Control -	SSWS Strainer BF509 Main Backwash Valve
1EA-HV-2197D	Control -	SSWS Strainer DF509 Main Backwash Valve
1EA-BP502	Control -	SSWS Pump BP502
1EA-HSS-2219B	Control -	Transfer Switch For SSWS Pump BP502
1EA-DP502	Control -	SSWS Pump DP502
1EA-HSS-2219D	Control -	Transfer Switch For SSWS Pump DP502

SSWS - REDUNDANT CONTROLS

1EA-HV-2203	Local Control -	RACS Heat Exchanger Supply Valve (From SACS Loop A)
1EA-AP502	Local Control -	SSWS Pump AP502
1EA-CP502	Local Control -	SSWS Pump CP502

CONTROL AREA CHILLED WATER SYSTEM (CACWS) - RSP

1GJ-BK400	Control -	Control Area Chiller BK400
1GJ-HSS-9652B	Control -	Transfer Switch For Control Area Chiller BK400
1GJ-BK403	Control -	Safety-Related Panel Room Chiller BK403
1GJ-HSS-9666B4	Control -	Transfer Switch For Safety-Related Panel Room Chiller BK403
1GJ-BP400	Control -	Control Area Chilled Water Circulating Pump BP400
1GJ-BP414	Control -	Safety-Related Panel Room Chilled Water Circulating Pump BP414

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEMS CONTROLS

CACWS - REDUNDANT CONTROLS

1GJ-AK400	Local Control -	Control Area Chiller AK400
1GJ-AK403	Local Control -	Safety-Related Panel Room Chiller AK403
1GJ-AP400	Local Control -	Control Area Chilled Water Circulating Pump AP400
1GJ-AP414	Local Control -	Safety-Related Panel Room Chilled Water Circulating Pump AP414

REACTOR RECIRCULATION SYSTEM -RSP

1BB-HV-F031B(3)	Indication -	Reactor Recirculation Pump BP201 Discharge Valve

SAFETY/RELIEF VALVES - RSP

1AB-PSV-F013F	Control -	Main Steam Line B Safety/Relief Valve
1AB-PSV-F013H	Control -	Main Steam Line D Safety/Relief Valve
1AB-PSV-F013M	Control -	Main Steam Line D Safety/Relief Valve

SAFETY/RELIEF VALVES - REDUNDANT CONTROLS

1AB-PSV-F013A	Local Control -	Main Steam Line A Safety/Relief Valve
1AB-PSV-F013E	Local Control -	Main Steam Line C Safety/Relief Valve

(1) The Remote Shutdown Panel (RSP) is Panel 10C399.
(2) Valve is signalled to open on RSP Takeover.
(3) Valve is signalled to close on RSP Takeover.
(4) Pump is signalled to run on RSP Takeover.
(5) Operation of valve 1EG-HV-2496B is ganged to operation of valve 1EG-HV-2522B.
(6) Operation of valve 1EG-HV-2496D is ganged to operation of valve 1EG-HV-2522D.
(7) Operation of valve 1EG-HV-2520B is ganged to operation of RHR Pump BP202.

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK [(a)]	CHANNEL CALIBRATION [(a)]
1. Reactor Vessel Pressure		
2. Reactor Vessel Water Level		
3. Safety/Relief Valve Position (Energization)		NA
4. Suppression Chamber Water Level		
5. Suppression Chamber Water Temperature		
6. RHR System Flow		
7. Safety Auxiliaries Cooling System Flow		
8. Safety Auxiliaries Cooling System Temperature		
9. RCIC System Flow		
10. RCIC Turbine Speed		
11. RCIC Turbine Bearing Oil Pressure Low Indication		
12. RCIC High Pressure/Low Pressure Turbine Bearing Temperature High Indication		

TABLE 4.3.7.4-1 (Continued)

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK [(a)]	CHANNEL CALIBRATION [(a)]
13. Condensate Storage Tank Level Low-Low Indication		
14. Standby Diesel Generator 1AG400 Breaker Indication		NA
15. Standby Diesel Generator 1BG400 Breaker Indication		NA
16. Standby Diesel Generator 1CG400 Breaker Indication		NA
17. Standby Diesel Generator 1DG400 Breaker Indication		NA
18. Switchgear Room Cooler 1AVH401 Status Indication		NA
19. Switchgear Room Cooler 1BVH401 Status Indication		NA
20. Switchgear Room Cooler 1CVH401 Status Indication		NA
21. Switchgear Room Cooler 1DVH401 Status Indication		NA

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2,3	80
2. Reactor Vessel Water level	2	1	1,2,3	80
3. Suppression Chamber Water level	2	1	1,2,3	80
4. Suppression Chamber Water Temperature*	2	1	1,2,3	80
5. Suppression Chamber Pressure	2	1	1,2,3	80
6. Drywell Pressure	2	1	1,2,3	80
7. Drywell Air Temperature	2	1	1,2,3	80
8. Deleted				
9. Safety/Relief Valve Position Indicators	2/valve**	1/valve**	1,2,3	80
10. Drywell Atmosphere Post-Accident Radiation Monitor	2	1	1,2,3	80
11. North Plant Vent Radiation Monitor #	1	1	1,2,3	81
12. South Plant Vent Radiation Monitor #	1	1	1,2,3	81
13. FRVS Vent Radiation Monitor #	1	1	1,2,3	81
14. Primary Containment Isolation Valve Position Indication ##	2/valve	1/valve	1,2,3	82

* Average bulk pool temperature.

** Acoustic monitoring and tail pipe temperature.

\# High range noble gas monitors.

\#\# One channel consists of the open limit switch, and the other channel consists of the closed limit switch.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel to OPERABLE status within 30 days, or immediately initiate actions in accordance with 6.9.2.

b. With the number of OPERABLE channels less than the Minimum Number of Channels shown in Table 3.3.7.5-1, (except for the Drywell Atmosphere Post Accident Radiation Monitor) restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. Deleted

d. With the number of OPERABLE Drywell Atmosphere Post Accident Radiation Monitor channels less than the Minimum Number of Channels requirement shown in Table 3.3.7.5-1, initiate action in accordance with ACTION 81, below.

ACTION 81 - With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, verify the valve(s) position by use of alternate indication methods. If the affected penetration is not isolated by either (i) a closed manual valve, (ii) a blind flange, or (iii) a deactivated automatic valve located outside primary containment, restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 82 -

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, verify the valves) position by use of alternate indication methods. If the affected penetration, is not isolated by either (i) a closed manual valve, (ii) a blind flange, or (iii) a deactivated automatic valve located outside primary containment, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK [(a)]	CHANNEL CALIBRATION [(a)]	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure			1,2,3
2. Reactor Vessel Water Level			1,2,3
3. Suppression Chamber Water Level			1,2,3
4. Suppression Chamber Water Temperature			1,2,3
5. Suppression Chamber Pressure			1,2,3
6. Drywell Pressure			1,2,3
7. Drywell Air Temperature			1,2,3
8. Deleted			
9. Safety/Relief Valve Position Indicators			1,2,3
10. Drywell Atmosphere Post-Accident Radiation Monitor		**	1,2,3
11. North Plant Vent Radiation Monitor#			1,2,3
12. South Plant Vent Radiation Monitor#			1,2,3
13. FRVS Vent Radiation Monitor#			1,2,3
14. Primary Containment Isolation Valve Position Indication			1,2,3

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

** CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

High range noble gas monitors.

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK:

a) In accordance with the Surveillance Frequency Control Program in CONDITION 2*, and

b) In accordance with the Surveillance Frequency Control Program in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** in accordance with the Surveillance Frequency Control Program.

b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3 cps with the detector fully inserted.

d. The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2* or 3 from OPERATIONAL CONDITION 1.

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* With IRM's on range 2 or below.

** Neutron detectors may be excluded from CHANNEL CALIBRATION.

3.3.7.7 DELETED

LIMITING CONDITION FOR OPERATION

3.3.7.8

The material originally contained in Section 3/4.3.7.8 was deleted with the issuance of the Full Power License. However, to maintain numerical continuity between the succeeding sections and existing station procedural references to those Technical Specifications Sections, 3/4.3.7.8 has been intentionally left blank.

PAGE 3/4 3-90 IS DELETED

INSTRUMENTATION

3.3.7.10 Deleted

3.3.7.11 Deleted

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3/4.3.8 DELETED

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3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.9.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

TABLE 3.3.9-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNIT	MININMUM OPERABLE CHANNELS	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Water Level-High, Level 8	3	1

TABLE 3.3.9-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level-High, Level 8	$\leq$ 54.0 inches*	$\leq$ 55.5 inches

*See Bases Figure B 3/4 3-1.

TABLE 4.3.9.1-1

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3/4.3.10 MECHANICAL VACUUM PUMP TRIP INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.10 Two channels of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 with mechanical vacuum pump in service and any main steam line not isolated.

ACTION:

a. With one channel of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip inoperable, restore the channel to OPERABLE status within 12 hours. Otherwise, trip the mechanical vacuum pumps, or isolate the main steam lines or be in HOT SHUTDOWN within the next 12 hours.

b. With mechanical vacuum pump trip capability not maintained:

   1. Trip the mechanical vacuum pumps within 12 hours; or

   2. Isolate the main steam lines within 12 hours; or

   3. Be in HOT SHUTDOWN within 12 hours.

c. When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into the associated ACTIONS may be delayed for up to 6 hours provided the mechanical vacuum pump trip capability is maintained.

SURVEILLANCE REQUIREMENTS

4.3.10 Each channel of the Main Steam Line Radiation - High, High function for the mechanical vacuum pump trip shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL CHECK in accordance with the Surveillance Frequency Control Program;

b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program;

c. Performance of a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program. The Allowable Value shall be ≤ 3.6 x normal background; and

d. Performance of a LOGIC SYSTEM FUNCTIONAL TEST, including mechanical vacuum pump trip breaker actuation, in accordance with the Surveillance Frequency Control Program.

3/4.3.11 OSCILLATION POWER RANGE MONITOR

LIMITING CONDITION FOR OPERATION

3.3.11 Four channels of the OPRM instrumentation shall be OPERABLE*. Each OPRM channel period based algorithm amplitude trip setpoint (Sp) shall be less than or equal to the Allowable Value as specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER.

ACTIONS

a. With one or more required channels inoperable:

1. Place the inoperable channels in trip within 30 days, or
2. Place associated RPS trip system in trip within 30 days, or
3. Initiate an alternate method to detect and suppress thermal hydraulic instability oscillations within 30 days.

b. With OPRM trip capability not maintained:

1. Initiate alternate method to detect and suppress thermal hydraulic instability oscillations within 12 hours, and
2. Restore OPRM trip capability within 120 days.

c. Otherwise, reduce THERMAL POWER to less than 24% RTP within 4 hours.

SURVEILLANCE REQUIREMENTS

4.3.11.1 Perform CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

4.3.11.2 Calibrate the local power range monitor in accordance with the Surveillance Frequency Control Program in accordance with Note f, Table 4.3.1.1-1 of TS 3/4.3.1.

4.3.11.3 Perform CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program. Neutron detectors are excluded.

4.3.11.4 Perform LOGIC SYSTEM FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

4.3.11.5 Verify OPRM is enabled when THERMAL POWER is ≥ 26.1% RTP and recirculation drive flow ≤ the value corresponding to the percentage of rated core flow as specified in the CORE OPERATING LIMITS REPORT in accordance with the Surveillance Frequency Control Program. The value specified in the CORE OPERATING LIMITS REPORT shall not be less than 60% of rated core flow.

4.3.11.6 Verify the RPS RESPONSE TIME is within limits in accordance with the Surveillance Frequency Control Program. Neutron detectors are excluded.

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* When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated ACTIONS may be delayed for up to 6 hours, provided the OPRM maintains trip capability.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

   a) Place the recirculation flow control system in the Local Manual mode, and
   b) Reduce THERMAL POWER to ≤ 60.86% of RATED THERMAL POWER, and
   c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit per Specification 2.1.2, and
   d) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
   e) Reduce the LINEAR HEAT GENERATION RATE (LHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
   f) Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and
   g) Perform surveillance requirement 4.4.1.1.2 if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow.

2. Within 4 hours, reduce the Average Power Range Monitor (APRM) Scram Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 2.2.1; otherwise, with the Trip Setpoints and Allowable Values associated with one trip system not reduced to those applicable for single recirculation loop operation, place the affected trip system in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specification 2.2.1.

3. Within 4 hours, reduce the APRM Control Rod Block Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 3.3.6; otherwise, with the Trip Setpoint and Allowable Values associated with one trip function not

*See Special Test Exception 3.10.4.

ACTION (Continued)

reduced to those applicable for single recirculation loop operation, place at least one affected channel in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specification 3.3.6.

4. Within 4 hours, reduce the Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 3.3.6; otherwise, with the Trip Setpoints and Allowable Values associated with one trip function not reduced to those applicable for single recirculation loop operation, place at least one affected channel in the tripped condition and within the following 6 hours, reduce the Trip setpoints and Allowable Values of the remaining channels to those applicable for single recirculation loop operation per Specification 3.3.6.

5. Deleted

6. Otherwise be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 With one reactor coolant system recirculation loop not in operation in accordance with the Surveillance Frequency Control Program verify that:

a. Reactor THERMAL POWER is ≤ 60.86% of RATED THERMAL POWER, and

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is less than or equal to 90% of rated pump speed.

4.4.1.1.2 With one reactor coolant system recirculation loop not in operation, within no more than 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements or Specifications 4.4.1.1.2b and 4.4.1.1.2c do not apply when the loop not in operation is isolated from the reactor pressure vessel.

4.4.1.1.3 DELETED.

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FIGURE 3.4.1.1-1

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REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS*

4.4.1.2 All jet pumps shall be demonstrated OPERABLE as follows:

a. Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 24% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when the recirculation pumps are operating in accordance with Specification 3.4.1.3.

   1. The indicated recirculation loop flow differs by more than 10% from the established pump speed-loop flow characteristics.
   2. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.
   3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from the established patterns by more than 20%.

b. During single recirculation loop operation, each of the above required jet pumps in the operating loop shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that no two of the following conditions occur:

   1. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established* pump speed-loop flow characteristics.
   2. The indicated total core flow differs by more than 10% from the established* total core flow value derived from single recirculation loop flow measurements.
   3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established* single recirculation loop pattern by more than 20%.

c. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 24% of RATED THERMAL POWER.

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* During startup following any refueling outage, baseline data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon conclusion of the baseline data analysis. Single loop baseline data shall be recorded the first time the unit enters single loop operation during an operating cycle.

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated core flow with effective core flow** greater than or equal to 70% of rated core flow.

b. 10% of rated core flow with effective core flow** less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* during two recirculation loop operation.

ACTION:

With the recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Declare the recirculation loop of the pump with the slower flow not in operation and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program.

* See Special Test Exception 3.10.4.

** Effective core flow shall be the core flow that would result if both recirculation loop flows were assumed to be at the smaller value of the two loop flows.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 13 of the following reactor coolant system safety/relief valves shall be OPERABLE*# with the specified code safety valve function lift settings:**

4 safety-relief valves @ 1108 psig ±3%
5 safety-relief valves @ 1120 psig ±3%
5 safety-relief valves @ 1130 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety valve function of two or more of the above listed fourteen safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open safety relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more of the above required safety/relief valve acoustic monitors inoperable, restore the inoperable monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*SRVs which perform as ADS function must also satisfy the OPERABILITY requirements of Specification 3.5.1, ECCS-Operating.

**The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

#SRVs which perform a low-low set function must also satisfy the OPERABILITY requirements of Specification 3.4.2.2, Safety/Relief Valves Low-Low Set Function.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be ≤ 30% of full open noise level by performance of a:

a. CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program, and a

b. CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

4.4.2.2 At least 1/2 of the safety relief valve pilot stage assemblies shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program, and they shall be rotated such that all 14 safety relief valve pilot stage assemblies are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program. All safety relief valves will be re-certified to meet a ±1% tolerance prior to returning the valves to service after setpoint testing.

4.4.2.3 The safety relief valve main (mechanical) stage assemblies shall be set pressure tested, reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint* (psig) ±2% Open	Close
F013H	1017	905
F013P	1047	935

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the relief valve function and/or the low-low set function of both of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system (excluding actual valve actuation) in accordance with the Surveillance Frequency Control Program.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The drywell floor and equipment drain sump monitoring system,
b. The drywell atmosphere gaseous radioactivity monitoring system,
c. All three of the following:
   1. The drywell air cooler condensate flow rate monitoring system,
   2. The drywell pressure monitoring system, and
   3. The drywell temperature monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the drywell floor and equipment drain sump monitoring system inoperable:

   1. operation may continue for 30 days provided that all monitoring systems in 3.4.3.1.b and 3.4.3.1.c are OPERABLE, and provided that preplanned manual calculation to quantify leak rate is performed at least once per four hours, or

   2. restore the system to OPERABLE status within 24 hours.

b. With the drywell atmosphere gaseous radioactivity monitoring system inoperable, operation may continue for 30 days provided that the monitoring systems required by 3.4.3.1.a and 3.4.3.1.c are OPERABLE, and provided that grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours.

c. With one monitoring system in 3.4.3.1.c inoperable, exert best efforts to restore the system to OPERABLE status within 30 days and if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause for the malfunction and plans for restoring the system to OPERABLE status.

   With two less than the number of monitoring systems required by 3.4.3.1.c OPERABLE, operation may continue for up to 30 days, provided that the drywell floor and equipment drain sump monitoring system in 3.4.3.1.a and the drywell atmosphere gaseous radioactivity monitoring system in 3.4.3.1.b are OPERABLE.

d. Otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Drywell atmosphere gaseous radioactivity monitoring system-performance of a CHANNEL CHECK, a CHANNEL FUNCTIONAL TEST and a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

b. The drywell pressure shall be monitored in accordance with the Surveillance Frequency Control Program and the drywell temperature shall be monitored in accordance with the Surveillance Frequency Control Program.

c. Drywell floor and equipment drain sump monitoring system-performance of a CHANNEL FUNCTIONAL TEST and a CHANNEL CALIBRATION TEST in accordance with the Surveillance Frequency Control Program.

d. Drywell air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST and a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm IDENTIFIED LEAKAGE averaged over any 24-hour period.

d. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1, at rated pressure.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any period of 24 hours or less.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual or deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any increase in UNIDENTIFIED LEAKAGE exceeding the limit in e above, implement preplanned leak location and isolation actions and either verify that the source of the leakage is not service-sensitive type 304 or 316 stainless steel or reduce the leakage rate-of-change to less than the limit within 4 hours or be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric gaseous radioactivity in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment drain sump flow rate in accordance with the Surveillance Frequency Control Program, and

c. Monitoring the drywell air coolers condensate flow rate in accordance with the Surveillance Frequency Control Program, and

d. Monitoring the drywell pressure in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage), and

e. Monitoring the reactor vessel head flange leak detection system in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage), and

f. Monitoring the drywell temperature in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to the IST Program and verifying the leakage of each valve to be within the specified limit:

a. In accordance with the Surveillance Frequency Control Program,** and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program.

** P.I.V. leak test extension to the first refueling outage is permissible for each RCS P.I.V. listed in Table 3.4.3.2-1, that is identified in Public Service Electric & Gas Company's letter to the NRC (letter No. NLR-N87047), dated April 3, 1987, as needing a plant outage to test. For this one time test interval, the requirements of Section 4.0.2 are not applicable.

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

1ST ISOLATION VALVE(S) NUMBERS(S)	2ND ISOLATION VALVE(S) NUMBER(S)	PRESSURE INDICATION	SERVICE
BE-V006 BE-V071	BE-V007	1-BE-PISH-N654A	'A' Core Spray/ HPCI Injection
BE-V002 BE-V072	BE-V003	1-BE-PISH-N654B	'B' Core Spray Injection
BC-V114 BC-V119	BC-V113	1-BC-PISH-N653A	'A' LPCI Injection
BC-V017 BC-V120	BC-V016	1-BC-PISH-N653B	'B' LPCI Injection
BC-V102 BC-V121	BC-V101	1-BC-PISH-N653C	'C' LPCI Injection
BC-V005 BC-V122	BC-V004	1-BC-PISH-N653D	'D' LPCI Injection
BC-V111 BC-V117	BC-V110	1-BC-PISH-N653A	'A' Shutdown Cooling Return to 'A' Recirc Loop
BC-V014 BC-V118	BC-V013	1-BC-PISH-N653B	'B' Shutdown Cooling Return to 'B' Recirc Loop
BC-V071	BC-V164	1-BC-PISH-N657	Shutdown Cooling Supply From 'B' Recirc Loop

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

SERVICE	INSTRUMENT	ALARM SETPOINT (psig)	ALARM ALLOWABLE VALUE (psig)
Core Spray	1-BE-PISH-N654A	475	≤500
Core Spray	1-BE-PISH-N654B	475	≤500
LPCI/RHR	1-BC-PISH-N653A	380	≤410
LPCI/RHR	1-BC-PISH-N653B	380	≤410
LPCI/RHR	1-BC-PISH-N653C	380	≤410
LPCI/RHR	1-BC-PISH-N653D	380	≤410
RHR	1-BC-PISH-N657	130	≤155

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3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries per gram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

a. Note: LCO 3.0.4.c is applicable.
In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant;

1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

2. Greater than $100/\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

c. In OPERATIONAL CONDITION 1 or 2, with:

1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour, or

2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

	TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1.	Gross Beta and Gamma Activity Determination	In accordance with the Surveillance Frequency Control Program	1, 2, 3
2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	In accordance with the Surveillance Frequency Control Program	1
3.	Radiochemical for $\bar{E}$ Determination	In accordance with the Surveillance Frequency Control Program *	1
4.	Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
		b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5.	Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	In accordance with the Surveillance Frequency Control Program	1

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

\# Until the specific activity of the primary coolant system is restored to within its limits.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (hydrostatic or leak testing), and Figure 3.4.6.1-2 (heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS), and Figure 3.4.6.1-3 (operations with a critical core other than low power PHYSICS TESTS), with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange metal temperature shall be maintained greater than or equal to 79°F when reactor vessel head bolting studs are under tension.

APPLICABILITY At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 as applicable, in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-3 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and in accordance with the Surveillance Frequency Control Program during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update the curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3.

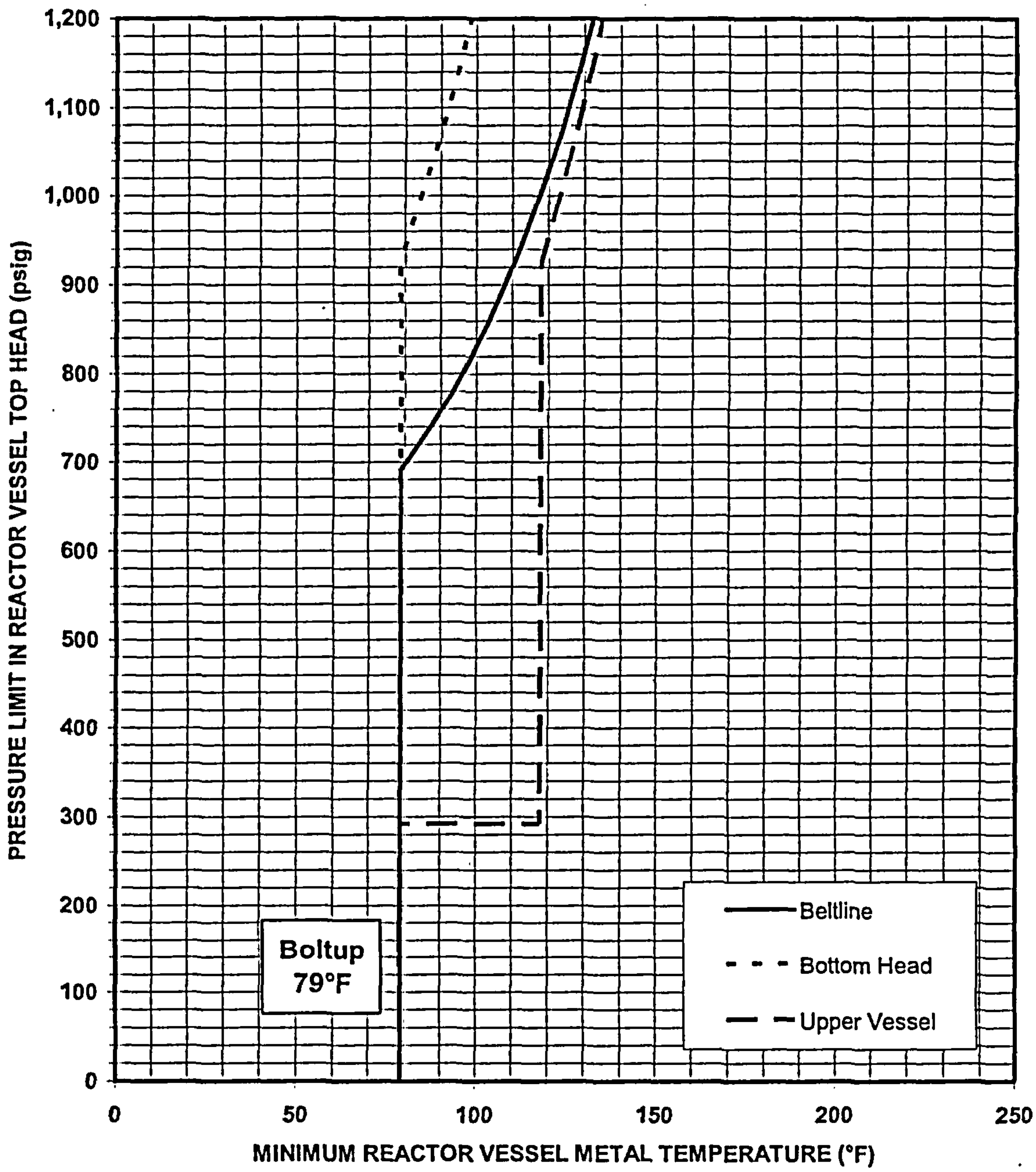
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to the limit specified in 3.4.6.1.d.

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 110°F, in accordance with the Surveillance Frequency Control Program.
   2. ≤ 90°F, in accordance with the Surveillance Frequency Control Program.

b. Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during tensioning of the reactor vessel head bolting studs.

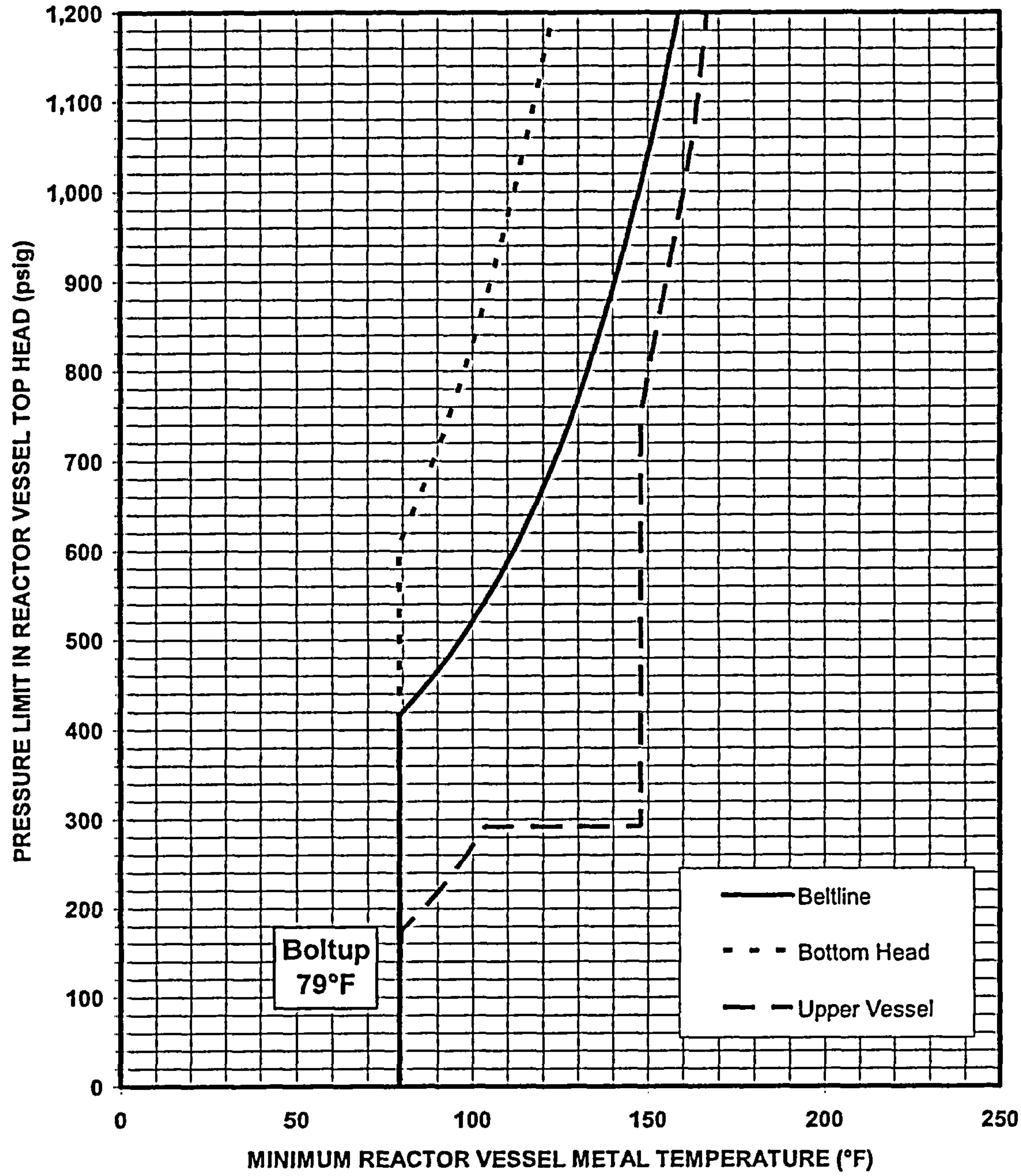
Figure 3.4.6.1-1
Hydrostatic Pressure and Leak Tests Pressure/Temperature Limits – Curve A



All system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with ASME Code Section XI.

This figure is valid for 32 EFPY of operation.

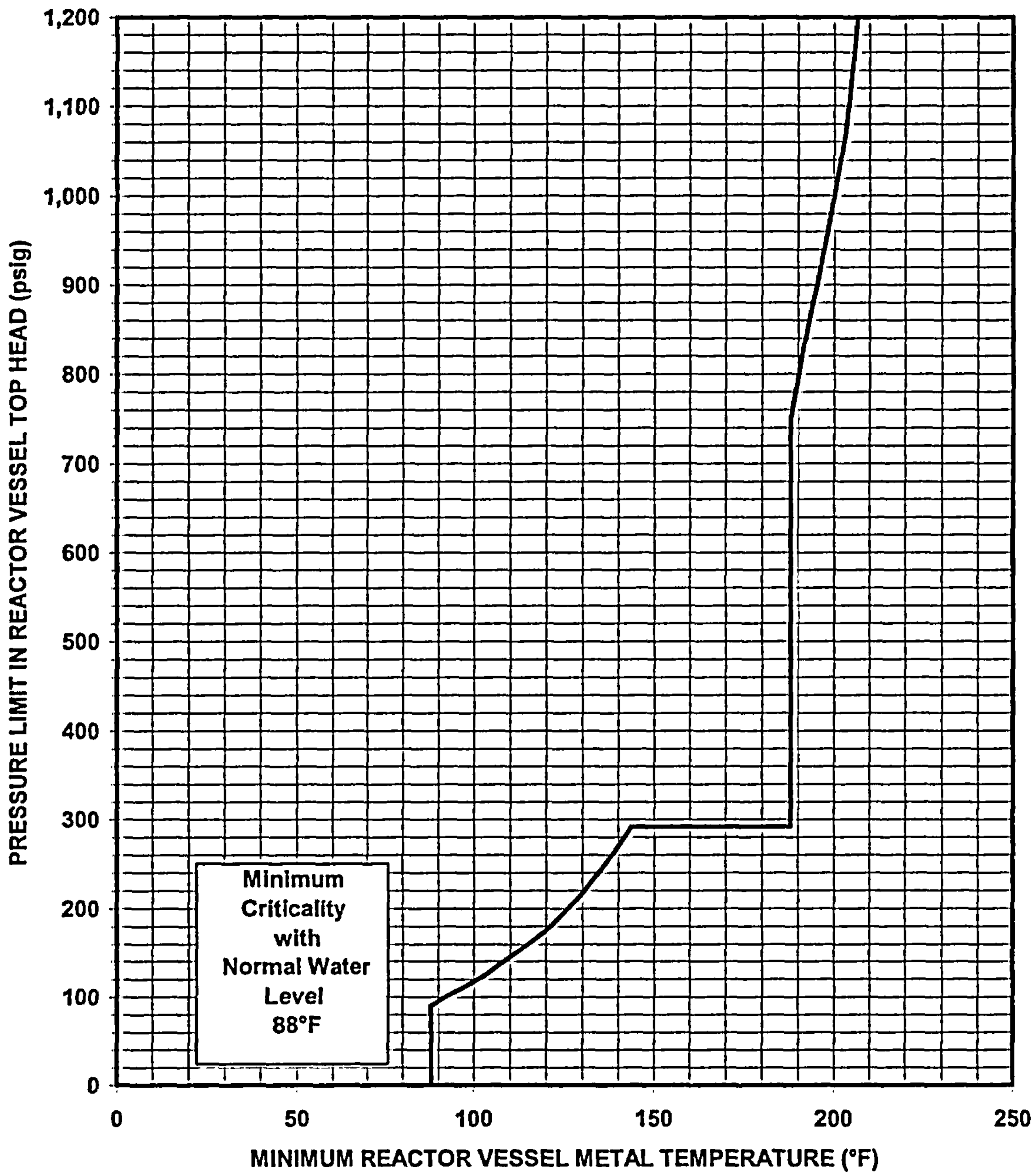
Figure 3.4.6.1-2
Non-Nuclear Heatup and Cooldown Pressure/Temperature Limits – Curve B



All heatup and cooldowns that are performed when the reactor is not critical at the normal heatup and cooldown rate.

This figure is valid for 32 EFPY of operation.

Figure 3.4.6.1-3
Core Critical Heatup and Cooldown Pressure/Temperature Limits – Curve C



All heatup and cooldowns that are performed when the reactor is critical at the normal heatup and cooldown rate.

This figure is valid for 32 EFPY of operation.

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REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1020 psig.

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1020 psig in accordance with the Surveillance Frequency Control Program.

* Not applicable during anticipated transients.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to the IST Program.

3/4.4.8 DELETED

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,##, with each loop consisting of:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Be in at least COLD SHUTDOWN within 24 hours.**

b. With no RHR shutdown cooling mode loop or recirculation pump in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least one shutdown cooling mode loop of the residual heat removal system, one recirculation pump, or alternate method shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation or at least one recirculation pump is in operation.

* The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

** Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.2 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,## with each loop consisting of:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4 and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 4.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop or recirculation pump in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system, recirculation pump or alternate method shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation or at least one recirculation pump is in operation.

* The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

The shutdown cooling mode loop may be removed from operation during hydrostatic testing.

** Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though COLD SHUTDOWN conditions are being maintained).

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

a. The core spray system (CSS) consisting of two subsystems with each subsystem comprised of:

   1. Two OPERABLE core spray pumps, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of four subsystems with each subsystem comprised of:

   1. One OPERABLE LPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. The high pressure coolant injection (HPCI) system consisting of:

   1. One OPERABLE HPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. The automatic depressurization system (ADS) with five OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*, **#, and 3*, **, ##.

---

*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

**The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

#See Special Test Exception 3.10.6.

##Two LPCI subsystems of the RHR system may be inoperable in that they are aligned in the shutdown cooling mode when the reactor vessel pressure is less than the RHR shutdown cooling permissive setpoint.

ACTION:

NOTE: LCO 3.0.4.b is not applicable to HPCI.

a. For the Core Spray system:

   1. With one core spray subsystem inoperable, provided that at least two LPCI subsystem are OPERABLE, restore the inoperable core spray subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. With both core spray subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. For the LPCI system:

   1. With one LPCI subsystem inoperable, provided that at least one core spray subsystem is OPERABLE, restore the inoperable LPCI subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. With two LPCI subsystems inoperable, provided that at least one core spray subsystem is operable, restore at least one LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   3. With three LPCI subsystems inoperable, provided that both core spray subsystems are OPERABLE, restore at least two LPCI subsystems to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   4. With all four LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

c. For the HPCI system, provided the Core Spray System, the LPCI system, the ADS and the RCIC system are OPERABLE:

* Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

1. With the HPCI system inoperable, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤200 psig within the following 24 hours.

2. With the HPCI system inoperable and either one LPCI subsystem or one CSS subsystem inoperable, restore the HPCI system to operable status within 72 hours or restore the LPCI subsystem/CSS subsystem to operable status within 72 hours. Otherwise, be in HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 200 psig in the next 24 hours.

d. For the ADS:

1. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤100 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤100 psig within the next 24 hours.

e. With a CSS and/or LPCI header ΔP instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or determine the ECCS header ΔP locally at least once per 12 hours; otherwise, declare the associated ECCS subsystem inoperable.

f. The discharge line "keep filled" alarm instrumentation associated with a LPCI and/or CSS subsystem(s) may be in an inoperable status for up to 6 hours for required surveillance testing* provided that the "keep filled" alarm instrumentation associated with at least one LPCI or CSS subsystem serviced by the affected "keep filled" system remains OPERABLE; otherwise, perform Surveillance Requirement 4.5.1.a.1.a.

g. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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* This includes testing of the "Reactor Coolant System Interface Valves Leakage Pressure Monitors" associated with LPCI and CSS in accordance with Surveillance 4.4.3.2.3

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. In accordance with the Surveillance Frequency Control Program:

1. For the core spray system, the LPCI system, and the HPCI system:

a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

b) Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

c) Verify the RHR System cross tie valves on the discharge side of the pumps are closed and power, if any, is removed from the valve operators.

2. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when tested pursuant to the IST Program:

1. The two core spray system pumps in each subsystem together develop a flow of at least 6150 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥105 psi above suppression pool pressure.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥20 psid.

3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of 1000 psig when steam is being supplied to the turbine at 1000, +20, -80 psig.**

c. In accordance with the Surveillance Frequency Control Program:

1. For the core spray system, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

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* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

2. For the HPCI system, verifying that:

   a) The system develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥200 psig, when steam is being supplied to the turbine at 200 + 15, -0 psig.**

   b) The suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - water level high signal.

3. Performing a CHANNEL CALIBRATION of the CSS, and LPCI system discharge line "keep filled" alarm instrumentation.

4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 4.4 psid.

5. Performing a CHANNEL CALIBRATION of the LPCI header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 1.0 psid.

d. For the ADS:

1. In accordance with the Surveillance Frequency Control Program, performing a CHANNEL FUNCTIONAL TEST of the Primary Containment Instrument Gas System low-low pressure alarm system.

2. In accordance with the Surveillance Frequency Control Program:

   a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

   b) Verify that when tested pursuant to the IST Program, that each ADS valve is capable of being opened.

   c) Performing a CHANNEL CALIBRATION of the Primary Containment Instrument Gas System low-low pressure alarm system and verifying an alarm setpoint of 85 ± 2 psig on decreasing pressure.

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. Core spray system subsystems with a subsystem comprised of:

1. Two OPERABLE core spray pumps, and

2. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

a) From the suppression chamber, or

b) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 135,000 available gallons of water.

b. Low pressure coolant injection (LPCI) system subsystems each with a subsystem comprised of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

a. With one of the above required subsystems inoperable, restore at least two subsystems to OPERABLE status within 4 hours or suspend all operations with a potential for draining the reactor vessel.

b. With both of the above required subsystems inoperable, suspend CORE ALTERATIONS and all operations with a potential for draining the reactor vessel. Restore at least one subsystem to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

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*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.

4.5.2.2 The core spray system shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2.a.2.b.

3/4.5.3 SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3 with an indicated water level of at least 74.5".

b. In OPERATIONAL CONDITION 4 and 5* with an indicated water level of at least 5.0" except that the suppression chamber level may be less than the limit or may be drained provided that:

1. No operations are performed that have a potential for draining the reactor vessel,
2. The reactor mode switch is locked in the Shutdown or Refuel position,
3. The condensate storage tank contains at least 135,000 available gallons of water, and
4. The core spray system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying the water level to be greater than or equal to:

a. 74.5" in accordance with the Surveillance Frequency Control Program in OPERATIONAL CONDITIONS 1, 2, and 3.

b. 5.0" in accordance with the Surveillance Frequency Control Program in OPERATIONAL CONDITIONS 4 and 5*.

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, in accordance with the Surveillance Frequency Control Program:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions * to be satisfied.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

b. In accordance with the Surveillance Frequency Control Program by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

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* See Special Test Exception 3.10.1

** Except valves, blind flanges, and deactivated automatic valves which are located inside the primary containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate (Type A test) in accordance with the Primary Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, other valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted.

c. *Less than or equal to 150 scfh per main steam line and less than or equal to 250 scfh combined through all four main steam lines when tested at 5 psig (leakage rate corrected to 1 Pa, 50.6 psig).

d. A combined leakage rate of less than or equal to 10 gpm for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines, when tested at 1.10 Pa, 55.7 psig.

e. A combined leakage rate of less than or equal to 10 gpm for all other penetrations and containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 Pa, 55.7 psig Δp.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:
With:

a. The measured overall integrated primary containment leakage rate (Type A test) not in accordance with the Primary Containment Leakage Rate Testing Program, or

b. The measured combined leakage rate exceeding the leakage rate specified in the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted, or

c. The measured leakage rate exceeding 150 scfh per main steam line or exceeding 250 scfh combined through all four main steam lines, or

*Exemption to Appendix "J" of 10 CFR 50.

ACTION (Continued)

d. The measured combined leakage rate for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines exceeding 10 gpm, or

e. The measured combined leakage rate for all other penetrations and containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 10 gpm,

restore:

a. The overall integrated leakage rate(s) (Type A test) to be in accordance with the Primary Containment Leakage Rate Testing Program, and

b. The combined leakage rate to be in accordance with the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted, and

c. The leakage rate to less than or equal to 150 scfh per main steam line and less than or equal to 250 scfh combined through all four main steam lines, and

d. The combined leakage rate for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines to less than or equal to 10 gpm, and

e. The combined leakage rate for all other penetrations and containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 10 gpm,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2.a The primary containment leakage rates shall be demonstrated in accordance with the Primary Containment Leakage Rate Testing Program for the following:

1. Type A test.

2. Type B and C tests (including air locks).

b. DELETED.

c. DELETED.

* Exemption to Appendix "J" of 10 CFR 50.

SURVEILLANCE REQUIREMENTS (Continued)

d. DELETED.

e. DELETED.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Containment isolation valves which form the boundry for the long-term seal of the feedwater lines shall be hydrostatically tested at 1.10 Pa, 55.7 psig, at least once per 18 months.

h. All containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

i. DELETED.

j. DELETED.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Primary Containment Leakage Rate Testing Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. By verifying seal leakage rate in accordance with the Primary Containment Leakage Rate Testing Program.

b. By conducting an overall air lock leakage test in accordance with the Primary Containment Leakage Rate Testing Program.

c. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.**

** Except that the inner door need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the inner door interlock is tested within 8 hours after the primary containment has been de-inerted.

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PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment shall be determined in accordance with the Primary Containment Leakage Rate Testing Program.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the containment, the inspection procedure, and the corrective actions taken.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between -0.5 and +1.5 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program:

	Elevation Zone	Approximate Azimuth*
a.	86'11"-112'8" (under vessel)	90°, 225°, 90°, 270°
b.	86'11"-111'10" (outside of pedestal)	135°, 300°, 100°, 190°
c.	111'10"-139'2"	55°, 240°, 155°, 315°
d.	139'2"-168'0"	45°, 215°, 0°, 90°,180°, 270°
e.	168'0"-192'7"	95°, 130°, 300°, 355°, 45°, 225°

* At least one reading from each elevation zone is required for a volumetric average calculation.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and suppression chamber purge system, including the 6-inch nitrogen supply line, may be in operation for up to 500 hours each 365 days with the supply and exhaust isolation valves in one supply line and one exhaust line open for containment prepurge cleanup, inerting, deinerting, or pressure control.*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With a drywell or suppression chamber purge supply and/or exhaust isolation valve and/or the nitrogen supply valve open, except as permitted above, close the valves(s) or otherwise isolate the penetration(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a drywell purge supply or exhaust isolation valve, or a suppression chamber purge supply or exhaust isolation valve or the nitrogen supply valve, having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.8.2, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8.1 Before being opened, the drywell and suppression chamber purge supply and exhaust, and nitrogen supply butterfly isolation valves shall be verified not to have been open for more than 500 hours in the previous 365 days.*

4.6.1.8.2 At least once per 24 months, the 26-inch drywell purge supply and exhaust isolation valves and the 24-inch suppression chamber purge supply and exhaust isolation valves and the 6-inch nitrogen supply valve shall be demonstrated OPERABLE in accordance with the Primary Containment Leakage Rate Testing Program.

* Valves open for pressure control are not subject to the 500 hours per 365 days limit, provided the 2-inch bypass lines are being utilized.

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. With an indicated water level between 74.5" and 78.5" and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

   a) 105°F during testing which adds heat to the suppression chamber.

   b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isolation valves closed following a scram.

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 0.80 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the suppression chamber average water temperature greater than 95°F and THERMAL POWER greater than 1% of RATED THERMAL POWER, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95° within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

LIMITING CONDITION FOR OPERATION (continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With one drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits in accordance with the Surveillance Frequency Control Program.

b. In accordance with the Surveillance Frequency Control Program in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than 95°F, by verifying:

a) Suppression chamber average water temperature to be less than or equal to 110°F.

c. At least once per 30 minutes in OPERATIONAL CONDITION 3 following a scram with suppression chamber average water temperature greater than 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

d. By an external visual examination of the suppression chamber after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 177°F and reactor coolant system pressure greater than 100 psig.

e. In accordance with the Surveillance Frequency Control Program by a visual inspection of the accessible interior and exterior of the suppression chamber.

SURVEILLANCE REQUIREMENTS (continued)

f. In accordance with the Surveillance Frequency Control Program by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 0.80 psi and verifying that the differential pressure does not decrease by more than 0.24 inch of water per minute for a period of 10 minutes. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the Surveillance Frequency Control Program schedule may be resumed.

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool spray sparger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 540 gpm on recirculation flow through the RHR heat exchanger (after consideration of flow through the closed bypass valve) and suppression pool spray sparger when tested pursuant to the IST Program.

* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 10,160 gpm on recirculation flow through the RHR heat exchanger (after consideration of flow through the closed bypass valve) and the suppression pool when tested pursuant to the IST Program.

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* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

CONTAINMENT SYSTEMS

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each primary containment isolation valve and each reactor instrumentation line excess flow check valve shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

   1. Restore the inoperable valve(s) to OPERABLE status, or

   2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

   3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.3 are not applicable provided that within 4 hours either:

   1. The inoperable valve is returned to OPERABLE status, or

   2. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

* Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shall be determined to be within its limit when tested pursuant to the IST Program.

4.6.3.4 In accordance with the Surveillance Frequency Control Program, verify that a representative sample of reactor instrumentation line excess flow check valves# actuates to the isolation position on a simulated instrument line break signal.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE*:

a. In accordance with the Surveillance Frequency Control Program by verifying the continuity of the explosive charge.

b. In accordance with the Surveillance Frequency Control Program by removing the explosive squib from at least one explosive valve, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

* Exemption to Appendix J of 10 CFR Part 50.

\# The reactor vessel head seal leak detection line (penetration J5C) is not required to be tested pursuant to this requirement.

TABLE 3.6.3-1

DELETED

Pages 3/4 6-20 through 3/4 6-42 have been intentionally omitted

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 All suppression chamber - drywell vacuum breakers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above vacuum breakers inoperable for opening restore the vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker not closed, close the open vacuum breaker within 2 hours; or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed in accordance with the Surveillance Frequency Control Program*.

b. Demonstrated OPERABLE:

1. In accordance with the Surveillance Frequency Control Program and within 12 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by performing a functional test of each vacuum breaker.

2. In accordance with the Surveillance Frequency Control Program by verifying the opening setpoint of each vacuum breaker to be less than or equal to 0.20 psid.

* Not required to be met for vacuum breaker assembly valves that are open during surveillances or that are open when performing their intended functions.

CONTAINMENT SYSTEMS

REACTOR BUILDING - SUPPRESSION CHAMBER VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Each reactor building - suppression chamber vacuum breaker assembly shall be OPERABLE

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one reactor building - suppression chamber vacuum breaker assembly, with one or two valves inoperable for opening, restore the vacuum breaker assembly to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With two reactor building - suppression chamber vacuum breaker assemblies with one or two valves inoperable for opening, restore both valves in one vacuum breaker assembly to OPERABLE status within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one or two reactor building - suppression chamber vacuum breaker assemblies, with one valve not closed, close the open vacuum breaker assembly valve(s) within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With two valves in one or two reactor building - suppression chamber vacuum breaker assemblies not closed, close one open vacuum breaker assembly valve in each affected assembly within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each reactor building - suppression chamber vacuum breaker assembly shall be:

a. Verified closed in accordance with the Surveillance Frequency Control Program*.

b. Demonstrated OPERABLE:

1. In accordance with the Surveillance Frequency Control Program by:

   a) Performing a functional test of each vacuum breaker assembly valve.

2. In accordance with the Surveillance Frequency Control Program by:

   a) Verifying the opening setpoint of each vacuum breaker assembly valve to be less than or equal to 0.25 psid.

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* Not required to be met for vacuum breaker assembly valves that are open during surveillances or that are open when performing their intended functions.

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3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying in accordance with the Surveillance Frequency Control Program that the reactor building is at a negative pressure.

b. Verifying in accordance with the Surveillance Frequency Control Program that:

 1. All secondary containment equipment hatches and blowout panels are closed and sealed.

 2. a. For double door arrangements, at least one door in each access to the secondary containment is closed.

    b. For single door arrangements, the door in each access to the secondary containment is closed except for routine entry and exit.

 3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

* When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS (Continued)

c. In accordance with the Surveillance Frequency Control Program:

1. Verifying that four filtration recirculation and ventilation system (FRVS) recirculation units and one ventilation unit of the filtration recirculation and ventilation system will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 375 seconds, and

2. Operating four filtration recirculation and ventilation system (FRVS) recirculation units and one ventilation unit of the filtration recirculation and ventilation system for four hours and maintaining greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate not exceeding 3324 CFM.

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment ventilation system (RBVS) automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable dampers to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated damper secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

Otherwise, in OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment ventilation system automatic isolation damper shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a Prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control or power circuit by cycling the damper through at least one complete cycle of full travel and verifying the specified isolation time.

b. In accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each isolation damper actuates to its isolation position.

c. By verifying the isolation time to be within its limit in accordance with the Surveillance Frequency Control Program.

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* When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

ISOLATION GROUP NO. 19

DAMPER FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Reactor Building Ventilation Supply Damper HD-9370A	7
2. Reactor Building Ventilation Supply Damper HD-9370B	7
3. Reactor Building Ventilation Exhaust Damper HD-9414A	7
4. Reactor Building Ventilation Exhaust Damper HD-9414B	7

CONTAINMENT SYSTEMS

3.6.5.3 FILTRATION, RECIRCULATION AND VENTILATION SYSTEM (FRVS)

FRVS VENTILATION SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3.1 Two FRVS ventilation units shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

a. With one of the above required FRVS ventilation units inoperable, restore the inoperable unit to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In Operational Condition *, place the OPERABLE FRVS ventilation unit in operation or suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both ventilation units inoperable in Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Each of the two ventilation units shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the water seal bucket traps have a water seal and making up any evaporative losses by filling the traps to the overflow.

b. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 15 minutes.

* When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS (continued)

c. In accordance with the Surveillance Frequency Control Program or upon determination** that the HEPA filters or charcoal adsorbent could have been damaged by structural maintenance or adversely affected by any chemicals, fumes or foreign materials (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rates are 9,000 cfm ± 10% for each FRVS ventilation unit.

2. Verifying within 31 days after removal from the FRVS ventilation units, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity 95%.

3. Verifying a subsystem flow rate of 9,000 cfm ± 10% for each FRVS ventilation unit during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the FRVS ventilation units, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

** This determination shall consider the maintenance performed and/or the type, quantity, length of contact time, known effects and previous accumulation history for all contaminants which could reduce the system performance to less than that verified by the acceptance criteria in items c.1 through c.3 below.

SURVEILLANCE REQUIREMENTS (continued)

e. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5 inches Water Gauge in the ventilation unit while operating the filter train at a flow rate of 9,000 cfm ± 10% for each FRVS ventilation unit.

2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

   a. Manual initiation from the control room, and

   b. Simulated automatic initiation signal.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2 March 1978, while operating the system at a flow rate of 9,000 cfm ± 10% for each FRVS ventilation unit.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9,000 cfm ± 10% for each FRVS ventilation unit.

3.6.5.3 FILTRATION, RECIRCULATION AND VENTILATION SYSTEM (FRVS)

FRVS RECIRCULATION SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3.2 Six FRVS recirculation units shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

a. With one or two of the above required FRVS recirculation units inoperable, restore all the inoperable unit(s) to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In Operational Condition*, suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With three or more of the above required FRVS recirculation units inoperable in Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

c. With three or more of the above required FRVS recirculation units inoperable in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3.2 Each of the six FRVS recirculation units shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the water seal bucket traps have a water seal and making up any evaporative losses by filling the traps to the overflow.

b. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and verifying that the subsystem operates for at least 15 minutes.

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* When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS (continued)

c. In accordance with the Surveillance Frequency Control Program or upon determination** that the HEPA filters could have been damaged by structural maintenance or adversely affected by any foreign materials (1) after any structural maintenance on the HEPA filters or housings by:

   1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rates are 30,000 cfm ± 10% for each FRVS recirculation unit.

   2. Verifying a subsystem flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit during system operation when tested in accordance with ANSI N510-1980.

d. not used

e. In accordance with the Surveillance Frequency Control Program by:

   1. Verifying that the pressure drop across the exhaust duct is less than 8 inches Water Gauge in the recirculation filter train while operating the filter train at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit.

   2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      a. Manual initiation from the control room, and

      b. Simulated automatic initiation signal.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2 March 1978, while operating the system at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit.

** This determination shall consider the maintenance performed and/or the type, quantity, length of contact time, known effects and previous accumulation history for all contaminants which could reduce the system performance to less than that verified by the acceptance criteria in items c.1 and c.2 below.

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CONTAINMENT SYSTEMS

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

CONTAINMENT HYDROGEN RECOMBINER SYSTEMS

The material originally contained in Section 3/4.6.6.1 was deleted with the issuance of Amendment No.160 . However, to maintain numerical continuity between the succeeding sections and existing station procedural references to those Technical Specification sections, 3/4.6.6.1 has been intentionally left blank.

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.2 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The drywell and suppression chamber oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program thereafter.

* See Special Test Exception 3.10.5.

3/4.7 PLANT SYSTEMS
3/4.7.1 SERVICE WATER SYSTEMS
SAFETY AUXILIARIES COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the following independent safety auxiliaries cooling system (SACS) subsystems, with each subsystem comprised of:

a. Two OPERABLE SACS pumps, and

b. An OPERABLE flow path consisting of a closed loop through the SACS heat exchangers and SACS pumps and to associated safety related equipment

shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, two subsystems.

b. In OPERATIONAL CONDITION 4, 5, and ** the subsystems associated with systems and components required OPERABLE by Specification 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1 and 3.9.11.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and **.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. a. With one SACS pump inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore the inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.*** If the condition specified by *** can not be met, be in at least HOT SHUTDOWN within the next 72 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one SACS heat exchanger inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore the heat exchanger to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN with the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one SACS subsystem otherwise inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, realign at least one of the affected diesel generators to the OPERABLE SACS subsystem within 2 hours, within 6 hours realign other affected SACS supported loads required to support plant operation for at least 72 hours, and restore the inoperable subsystem to OPERABLE status with at least one OPERABLE pump and heat exchanger within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.***

** When handling recently irradiated fuel in the secondary containment.
*** Two diesel generators and two service water pumps associated with the unaffected SACS loop must be OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. a. With one SACS pump in each subsystem inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore at least one inoperable pump to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.***

   b. With one SACS heat exchanger in each subsystem inoperable, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With both SACS subsystems otherwise inoperable, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* in the following 24 hours.

b. In OPERATIONAL CONDITION 3 or 4 with the SACS subsystem, which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, having two SACS pumps or one heat exchanger inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

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* Whenever both SACS subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

*** Two diesel generators and service water pumps associated with the required OPERABLE SACS pumps and all SACS heat exchangers must be OPERABLE.

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LIMITING CONDITION FOR OPERATION (continued)

ACTION: (Continued)

c. In OPERATIONAL CONDITION 4 or 5 with the SACS subsystem, which is associated with safety related equipment required OPERABLE by Specification 3.5.2, having two SACS pumps or one heat exchanger inoperable, declare the associated safety related equipment inoperable and take the ACTION required by Specification 3.5.2.

d. In OPERATIONAL CONDITION 5 with the SACS subsystem, which is associated with an RHR loop required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, having two SACS pumps or one heat exchanger inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

e. In OPERATIONAL CONDITION 4, 5, or **, with one SACS subsystem, which is associated with safety related equipment required OPERABLE by Specification 3.8.1.2, inoperable, realign the associated diesel generators within 2 hours to the OPERABLE SACS subsystem, or declare the associated diesel generators inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

f. In OPERATIONAL CONDITION 4, 5, or **, with only one SACS pump and heat exchanger and its associated flowpath OPERABLE, restore at least two pumps and two heat exchangers and associated flowpaths to OPERABLE status within 72 hours or, declare the associated safety related equipment inoperable and take the associated ACTION requirements.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required safety auxiliaries cooling system subsystems shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program by verifying that: 1) Each automatic valve servicing safety-related equipment actuates to its correct position on the appropriate test signal(s), and 2) Each pump starts automatically when its associated diesel generator automatically starts.

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least the following independent station service water system loops, with each loop comprised of:

a. Two OPERABLE station service water pumps, and
b. An OPERABLE flow path capable of taking suction from the Delaware River (ultimate heat sink) and transferring the water to the SACS heat exchangers,

shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, two loops.
b. In OPERATIONAL CONDITION 4, 5 and *, one loop.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one station service water pump inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore the inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.** If the condition specified by ** can not be met, be in at least HOT SHUTDOWN within the next 72 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one station service water pump in each loop inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, restore at least one inoperable pump to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.***

3. With one station service water system loop otherwise inoperable, and if continued plant operation is permitted by LCO 3.7.1.3, assess the operability of the associated SACS loop and take the ACTION specified in LCO 3.7.1.1, Action Statement a.2, if required, and restore the inoperable station service water system loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.**

* When handling recently irradiated fuel in the secondary containment.
** Two diesel generators and two SACS pumps associated with the unaffected service water loop must be OPERABLE.
*** Two diesel generators and SACS pumps associated with the required OPERABLE service water pumps and all SACS heat exchangers must be OPERABLE.

LIMITING CONDITION FOR OPERATION (continued)

ACTION: (Continued)

b. In OPERATIONAL CONDITION 4 or 5:

With only one station service water pump and its associated flowpath OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated SACS subsystem inoperable and take the ACTION required by Specification 3.7.1.1.

c. In OPERATIONAL CONDITION *:

With only one station service water pump and its associated flowpath OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated SACS subsystem inoperable and take the ACTION required by Specification 3.7.1.1. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 At least the above required station service water system loops shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic), servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program, by verifying that:

1. Each automatic valve servicing non-safety related equipment actuates to its isolation position on an isolation test signal.

2. Each pump starts automatically when its associated diesel generator automatically starts.

* When handling recently irradiated fuel in the secondary containment.

PLANT SYSTEMS

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The ultimate heat sink (Delaware River) shall be OPERABLE with:

a. A minimum river water level at or above elevation -9'0 Mean Sea Level, USGS datum (80'0 PSE&G datum), and

b. An average river water temperature of less than or equal to 85.0°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

With the river water temperature in excess of 85.0°F, continued plant operation is permitted provided that both emergency discharge valves are open and emergency discharge pathways are available. With the river water temperature in excess of 88.0°F, continued plant operation is permitted provided that all of the following additional conditions are satisfied: all SSWS pumps are OPERABLE, all SACS pumps are OPERABLE, all EDGs are OPERABLE and the SACS loops have no cross-connected loads (unless they are automatically isolated during a LOP and/or LOCA); with ultimate heat sink temperature greater than 89°F and less than or equal to 91.4°F, verify once per hour that water temperature of the ultimate heat sink is less than or equal to 89°F averaged over the previous 24 hour period; otherwise, with the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITIONS 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITIONS 4 or 5, declare the SACS system and the station service water system inoperable and take the ACTION required by Specification 3.7.1.1 and 3.7.1.2.

c. In Operational Condition *, declare the plant service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The ultimate heat sink shall be determined OPERABLE:

a. By verifying the river water level to be greater than or equal to the minimum limit in accordance with the Surveillance Frequency Control Program.

b. By verifying river water temperature to be within its limit:

   1) in accordance with the Surveillance Frequency Control Program when the river water temperature is less than or equal to 82°F.

   2) in accordance with the Surveillance Frequency Control Program when the river water temperature is greater than 82°F.

* When handling recently irradiated fuel in the secondary containment.

PLANT SYSTEMS

3/4.7.2 CONTROL ROOM SYSTEMS

CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2.1 Two control room emergency filtration system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3

1. With one control room emergency filtration subsystem inoperable for reasons other than Condition a.2, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one or more control room emergency filtration subsystems inoperable due to an inoperable control room envelope (CRE) boundary##,

a. Immediately, initiate action to implement mitigating actions; and

b. Within 24 hours, verify mitigating actions ensure CRE occupant exposures to radiological and chemical hazards will not exceed the limits and actions to mitigate exposure to smoke hazards are taken; and

c. Within 90 days, restore the CRE boundary to operable status;

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION *:

1. With one control room emergency filtration subsystem inoperable for reasons other than Condition b.3, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the pressurization/recirculation mode of operation.

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* When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

\## The main control room envelope (CRE) boundary may be opened intermittently under administrative control.

CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION (continued)

2. With both control room emergency filtration subsystems inoperable for reasons other than Condition b.3, suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. With one or more control room emergency filtration subsystems inoperable due to an inoperable CRE boundary##, immediately suspend handling of recently irradiated fuel and operations with a potential for draining the vessel.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION *.

SURVEILLANCE REQUIREMENTS

4.7.2.1.1 Each control room emergency filtration subsystem shall be demonstrated OPERABLE:

a. DELETED

b. In accordance with the Surveillance Frequency Control Program by verifying that the subsystem operates for at least 10 hours with the heaters on.

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* When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

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

SURVEILLANCE REQUIREMENTS (Continued)

c. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem filter train by:

   1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system filter train flow rate is 4000 cfm ± 10%.

   2. Verifying within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 0.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity 70%.

   3. Verifying a subsystem filter train flow rate of 4000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the Control Room Emergency Filtration units that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 0.5% when tested in accordance with ATSM D3803 –1989 at a temperature of 30°C and a relative humidity of 70%.

e. In accordance with the Surveillance Frequency Control Program by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.5 inches Water Gauge while operating the filter train subsystem at a flow rate of 4000 cfm ± 10%.

   2. Verifying with the control room hand switch in the recirculation mode that on each of the below recirculation mode actuation test signals, the subsystem automatically switches to the isolation mode of operation and the isolation dampers close within 5 seconds:

      a) High Drywell Pressure
      b) Reactor Vessel Water Level Low Low Low, Level 1
      c) Control room ventilation radiation monitors high.

3. Verifying with the control room hand switch in the outside air mode that on each of the below pressurization mode actuation test signals, the subsystem automatically switches to the pressurization mode of operation:

   a) High Drywell Pressure
   b) Reactor Vessel Water Level Low Low Low, Level 1
   c) Control room ventilation radiation monitors high.

4. Verifying that the heaters dissipate 13 ± 1.3 Kw when tested in accordance with ANSI N510-1980 and verifying humidity is maintained less than or equal to 70% humidity through the carbon adsorbers by performance of a channel calibration of the humidity control instrumentation.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the system at a flow rate of 4000 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

4.7.2.1.2 The control room envelope boundary shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Control Room Envelope Habitability Program by performance of control room envelope unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

PLANT SYSTEMS

CONTROL ROOM AIR CONDITIONING (AC) SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2.2 Two control room AC subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3:

   1. With one control room AC subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. With two control room AC subsystems inoperable:

      a. Verify control room air temperature is less than 90°F at least once per 4 hours; and

      b. Restore one control room AC subsystem to OPERABLE status within 72 hours.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION *:

   1. With one control room AC subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days; or place the OPERABLE control room AC subsystem in operation; or immediately suspend movement of recently irradiated fuel assemblies in the secondary containment and initiate action to suspend operations with a potential for draining the reactor vessel.

   2. With two control room AC subsystems inoperable, immediately suspend movement of recently irradiated fuel assemblies in the secondary containment and initiate action to suspend operations with a potential for draining the reactor vessel.

   3. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

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* When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

CONTROL ROOM AIR CONDITIONING (AC) SYSTEM

SURVEILLANCE REQUIREMENTS

4.7.2.2 Each control room AC subsystem shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying each subsystem has the capability to remove the assumed heat load.

3/4.7.3 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.3 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Delaware River reaches 6.0 feet Mean Sea Level (MSL) USGS datum (95.0 feet PSE&G datum) at the Service Water Intake Structure.

APPLICABILITY: At all times.

ACTION:

a. With severe storm warnings from the National Weather Service which may impact Artificial Island in effect or with the water level at the service water intake structure above elevation 6.0 feet MSL USGS datum (95.0 feet PSE&G datum), initiate and complete:

   1. The closing of all service water intake structure watertight perimeter flood doors identified in Table 3.7.3-1 within 1 hour, or declare affected service water system components inoperable and take the actions required by LCO 3.7.1.2;

   - and -

   2. The closing of all power block watertight perimeter flood doors identified in Table 3.7.3-1 within 1.5 hours.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Once closed, all access through the doors shall be administratively controlled.

b. With the water level at the service water intake structure above elevation 10.5 feet MSL USGS datum (99.5 feet PSE&G datum), be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The water level at the service water intake structure shall be determined to be within the limit by:

a. Measurement in accordance with the Surveillance Frequency Control Program when the water level is below elevation 6.0 MSL USGS datum (95.0 feet PSE&G datum), and

b. Measurement at least once per 4 hours when severe storm warnings from the National Weather Service which may impact Artificial Island are in effect.

c. Measurement in accordance with the Surveillance Frequency Control Program when the water level is equal to or above elevation 6.0 MSL USGS datum (95.0 feet PSE&G datum).

TABLE 3.7.3-1

PERIMETER FLOOD DOORS

INTAKE STRUCTURE DOORS

Water tight door 1
Water tight door 3
Water tight door 6
Water tight door 8

POWER BLOCK DOORS and HATCH

Doors & Hatch		Location
Hatch	Exterior	45; K
S-13	"	45.5; L
3340B	"	44; M
3337B	"	44; Md
6312	"	45.4; T
6323B	"	45.4; U
5315A	"	29.9; X
5315C	"	29; X
4323A	"	13.6; U
4304	"	13.6; U
3301A	"	13.6; Md
3305B	"	13.6; L
3315B	Interior-102'	25; H
3329A	"	27; H
3331B	"	35; H
3209A	Interior	26; H

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

Note: LCO 3.0.4.b is not applicable to RCIC.

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

   1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   2. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

   3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to the IST Program by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

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* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

SURVEILLANCE REQUIREMENTS (continued)

c. In accordance with the Surveillance Frequency Control Program by:

1. Performing a system functional test which includes simulated automatic actuation and restart# and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded.

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, - 0 psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.

* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

\# Automatic restart on a low water level signal which is subsequent to a high water level trip.

PLANT SYSTEMS

3/4.7.5 DELETED

Pages 3/4 7-14 through 3/4 7-18 have been intentionally omitted.

3/4.7.6 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.6 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.6.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in use - In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3, and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.6.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

3/4.7.7 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER.

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 2 hours or reduce THERMAL POWER to less than or equal to 24% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 The main turbine bypass system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. In accordance with the Surveillance Frequency Control Program by:

   1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

   2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME meets the following requirements when measured from the initial movement of the main turbine stop or control valve:

      a) 80% of turbine bypass system capacity shall be established in less than or equal to 0.3 second.

      b) Bypass valve opening shall start in less than or equal to 0.1 second.

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Four separate and independent diesel generators, each with:

   1. A separate fuel oil day tank containing a minimum of 360 gallons of fuel,

   2. A separate fuel storage system consisting of two storage tanks containing a minimum of 44,800 gallons of fuel, and

   3. A separate fuel transfer pump for each storage tank.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Note: LCO 3.0.4.b is not applicable to DGs.

a. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the inoperable offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one diesel generator of the above required A.C. electrical power sources inoperable,

   1. Demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 separately for each diesel generator within 24 hours* unless the absence of any potential common mode failure for the remaining diesel generators is demonstrated.

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* This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. For the inoperable A or B diesel generator, if continued operation is permitted by LCO 3.7.1.3:

   a) Restore the inoperable diesel generator to OPERABLE status within 72 hours, or

   b) Verify the Salem Unit 3 gas turbine generator (GTG) is available within 72 hours and once per 12 hours thereafter#, and restore the inoperable diesel generator to OPERABLE status within 14 days.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. For the inoperable C or D diesel generator, if continued operation is permitted by LCO 3.7.1.3, restore the inoperable diesel generator to OPERABLE status within 14 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one offsite circuit of the above required A.C. sources and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If a diesel generator became inoperable due to any causes other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators separately for each diesel generator by performing Surveillance Requirement 4.8.1.1.2.a.4 within 16 hours unless the absence of any potential common mode failure for the remaining diesel generators is demonstrated*. If continued operation is permitted by LCO 3.7.1.3, restore at least two offsite circuits and all four of the above required diesel generators to OPERABLE status within 72 hours from time of the initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. A successful test(s) of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 performed under this ACTION statement for the OPERABLE diesel generators satisfies the diesel generator test requirements of ACTION Statement b.

d. With both of the above required offsite circuits inoperable, restore at least one of the above required offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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* This test is required to be completed regardless of when the inoperable diesel generator is restored, to OPERABILITY.

\# After the initial verification period, the GTG may be unavailable for a single period of up to 24-hours and the once-per 12-hour requirement to verify that the GTG is available may be suspended during this period.

ACTION: (Continued)

e. With two diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either of the diesel generators became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 separately for each diesel generator within 8 hours* unless the absence of any potential common mode failure for the remaining diesel generators is demonstrated. Restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. A successful test(s) of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 performed under this ACTION statement for the OPERABLE diesel generators satisfies the diesel generator test requirements of ACTION Statement b.

f. With two diesel generators of the above required A.C. electrical power sources inoperable, in addition to ACTION e., above, verify within 2 hours that all required systems, subsystems, trains, components, and devices that depend on the remaining diesel generators as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

g. With one offsite circuit and two diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either of the diesel generators became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 separately for each diesel generator within 8 hours* unless the absence of any potential common mode failure for the remaining diesel generators is demonstrated. Restore at least one of the above required inoperable A.C. sources to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore the inoperable offsite circuit and both of the inoperable diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

h. With the buried fuel oil transfer piping's cathodic protection system inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the system to OPERABLE status.

i. With one fuel oil transfer pump inoperable, realign the flowpath of the affected tank to the tank with the remaining operable fuel oil transfer pump within 48 hours and restore the inoperable transfer pump to OPERABLE status within 14 days, otherwise declare the affected emergency diesel generator (EDG) inoperable. This variance may be applied to only one EDG at a time.

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* This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during shutdown by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE: *

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the fuel level in the fuel oil day tank.

2. Verifying the fuel level in the fuel oil storage tank.

3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the fuel oil day tank.

4. Verifying each diesel generator starts** from standby conditions and achieves steady state voltage $\geq$ 3828 and $\leq$ 4580 volts and frequency of 60 $\pm$ 1.2 Hz.

5. Verifying the diesel generator is synchronized, loaded to between 4000 and 4400*** kw and operates with this load for at least 60 minutes.

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* All engine starts and loading for the purpose of this surveillance testing may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

** A modified diesel generator start involving idling and gradual acceleration to synchronous speed may be used for this surveillance. When modified start procedures are not used, the time, voltage, and frequency tolerances of Surveillance Requirement 4.8.1.1.2.g must be met.

*** Momentary transients outside the load range do not invalidate this test.

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to 325 psig.

8. Verifying the lube oil pressure, temperature and differential pressure across the lube oil filters to be within manufacturer's specifications.

b. In accordance with the Surveillance Frequency Control Program by visually examining a sample of lube oil from the diesel engine to verify absence of water.

c. In accordance with the Surveillance Frequency Control Program and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the fuel oil day tank.

d. In accordance with the Surveillance Frequency Control Program by removing accumulated water from the fuel oil storage tanks.

e. In accordance with the Surveillance Frequency Control Program by performing a functional test on the emergency load sequencer to verify operability.

f. In accordance with the surveillance interval specified in the Diesel Fuel Oil Testing Program and prior to the addition of new fuel oil to the storage tank, samples shall be taken to verify fuel oil quality. Sampling and testing of new and stored fuel oil shall be in accordance with the Diesel Fuel Oil Testing Program contained in Specification 6.8.4.e.

SURVEILLANCE REQUIREMENTS (Continued)

g. In accordance with the Surveillance Frequency Control Program by verifying each diesel generator starts from standby conditions and achieves ≥ 3950 volts and ≥ 58.8 Hz in ≤ 10 seconds after receipt of the start signal, and subsequently achieves steady state voltage ≥ 3828 and ≤ 4580 volts and frequency of 60 ± 1.2 Hz.

h. In accordance with the Surveillance Frequency Control Program#, during shutdown, by:

1. Deleted.

2. Verifying the diesel generator capability to reject a load of greater than or equal to that of the RHR pump motor for each diesel generator while maintaining voltage ≥ 3828 and ≤ 4580 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the diesel generator capability to reject a load of 4430 kW without tripping. The generator voltage shall not exceed 4785 volts during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

a) Verifying loss of power is detected and deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds after receipt of the start signal, energizes the autoconnected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained ≥ 3828 and ≤ 4580 volts and 60 ± 1.2 Hz during this test.

For any start of a diesel generator, the diesel may be loaded in accordance with the manufacturer's recommendations.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The diesel generator shall achieve ≥ 3950 volts and ≥ 58.8 Hz in ≤ 10 seconds following receipt of the start signal and subsequently achieve steady state voltage ≥ 3828 and ≤ 4580 volts and frequency of 60 ± 1.2 Hz.

6. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

   a) Verifying loss of power is detected and deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds after receipt of the start signal, energizes the autoconnected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained ≥ 3828 and ≤ 4580 volts and 60 ± 1.2 Hz during this test.

7. Verifying that all automatic diesel generator trips, except engine overspeed, generator differential current, generator overcurrent, bus differential current and low lube oil pressure are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal.#

8. Deleted.

9. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 4430 kW.

Generator differential current, generator overcurrent, and bus differential current is two-out-of-three logic and low lube oil pressure is two-out-of-four logic.

SURVEILLANCE REQUIREMENTS (Continued)

10. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source,

    c) Be restored to its standby status, and

    d) Diesel generator circuit breaker is open.

11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

12. Verifying that the fuel oil transfer pump transfers fuel oil from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

13. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

14. Deleted.

i. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting all diesel generators simultaneously, during shutdown, and verifying that all diesel generators accelerate to at least 514 rpm in less than or equal to 10 seconds.

j. In accordance with the Surveillance Frequency Control Program by:

   1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or equivalent, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI Article IWD-5000.

k. In accordance with the Surveillance Frequency Control Program# by:

1. Verifying the diesel generator operates for at least 24 hours. During the first 22 hours of this test, the diesel generator shall be loaded to between 4000 and 4400 kW## and during the remaining 2 hours of this test, the diesel generator shall be loaded to between 4652 and 4873 kW. The diesel generator shall achieve ≥ 3950 volts and ≥ 58.8 Hz in ≤ 10 seconds following receipt of the start signal and subsequently achieve steady state voltage ≥ 3828 and ≤ 4580 volts and frequency of 60 ± 1.2 Hz.

2. Within 5 minutes after completing 4.8.1.1.2.k.1, verify each diesel generator starts and achieves ≥ 3950 volts and ≥ 58.8 Hz in ≤ 10 seconds after receipt of the start signal, and subsequently achieves steady state voltage ≥ 3828 and ≤ 4580 volts and frequency of 60 ± 1.2 Hz.

   - OR -

   Operate the diesel generator between 4000 kW and 4400 kW for two hours. Within 5 minutes of shutting down the diesel generator, verify each diesel generator starts and achieves ≥ 3950 volts and ≥ 58.8 Hz in ≤ 10 seconds after receipt of the start signal, and subsequently achieves steady state voltage ≥ 3828 and ≤ 4580 volts and frequency of 60 ± 1.2 Hz. This test shall continue for at least five minutes.

4.8.1.1.3 Reports - Not used.

4.8.1.1.4 The buried fuel oil transfer piping's cathodic protection system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by subjecting the cathodic protection system to a performance test.

For any start of a diesel generator, the diesel may be loaded in accordance with manufacturer's recommendations.

Momentary transients outside the load range do not invalidate this test.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Not used

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators, one of which shall be diesel generator A or diesel generator B, each with:

1. A separate fuel oil day tank containing a minimum of 360 gallons of fuel.
2. A fuel storage system consisting of two storage tanks containing a minimum of 44,800 gallons of fuel.
3. A separate fuel transfer pump for each storage tank.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of recently irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22'-2" above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

c. With one fuel oil transfer pump inoperable, realign the flowpath of the affected tank to the tank with the remaining operable fuel oil transfer pump within 48 hours and restore the inoperable transfer pump to OPERABLE status within 14 days, otherwise declare the affected emergency diesel generator (EDG) inoperable. This variance may be applied to only one EDG at a time.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

* When handling recently irradiated fuel in the secondary containment.

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Channel A, consisting of:
   1. 125 volt battery 1AD411
   2. 125 volt full capacity charger 1AD413 or 1AD414
   3. 250 volt battery 10D421;
   4. 250 volt full capacity charger 10D423

b. Channel B, consisting of:
   1. 125 volt battery 1BD411
   2. 125 volt full capacity charger 1BD413 or 1BD414
   3. 250 volt battery 10D431;
   4. 250 volt full capacity charger 10D433

c. Channel C, consisting of:
   1. 125 volt battery 1CD411
   2. 125 volt full capacity charger 1CD413 or 1CD414
   3. 125 volt battery 1CD447
   4. 125 volt full capacity charger 1CD444

d. Channel D, consisting of:
   1. 125 volt battery 1DD411
   2. 125 volt full capacity charger 1DD413 or 1DD414
   3. 125 volt battery 1DD447
   4. 125 volt full capacity charger 1DD444

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any 125v battery and/or all associated chargers of the above required D.C. electrical power sources inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With any 250v battery and/or charger of the above required DC electrical power sources inoperable, declare the associated HPCI or RCIC system inoperable and take the appropriate ACTION required by the applicable Specification.

LIMITING CONDITION FOR OPERATION (Continued)

c. With the average electrolyte temperature of each sixth cell of connected cells in any 125v battery at or below 72°F, but at or above 65°F, the battery may be considered OPERABLE for an additional 31 days, provided that:

1. Within 2 hours from identification of degraded temperature, the battery pilot cells are determined to meet Category A limits; and

2. Within 24 hours from identification of degraded temperature, and once per seven days thereafter, all connected cells are determined to meet Category B limits.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the average electrolyte temperature of each sixth cell of connected cells in any 250v battery at or below 72°F, but at or above 65°F, the battery may be considered OPERABLE for an additional 31 days, provided that:

1. Within 2 hours from identification of degraded temperature, the battery pilot cells are determined to meet Category A limits; and

2. Within 24 hours from identification of degraded temperature, and once per seven days thereafter, all connected cells are determined to meet Category B limits.

Otherwise, declare the associated HPCI or RCIC system inoperable and take the appropriate ACTION required by the applicable Specification.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required batteries and chargers shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and
2. Total battery terminal voltage for each 125-volt battery is greater than or equal to 129 volts on float charge and for each 250-volt battery the terminal voltage is greater than or equal to 258 volts on float charge.

b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 108 volts for a 125-volt battery or 210 volts for a 250-volt battery, or battery overcharge with battery terminal voltage above 140 volts for a 125-volt battery or 280 volts for a 250-volt battery, by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x 10^{-6} ohms, excluding cable intercell connections, and

3. The average electrolyte temperature of each sixth cell of connected cells is above 72°F.

c. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150 x 10^{-6} ohms, excluding cable intercell connections, and

4. The battery charger will supply the current listed below at the voltage listed below for at least 8 hours.

CHARGER	Minimum Voltage	CURRENT (AMPERES)
1AD413, 1AD414 1BD413, 1BD414 1CD413, 1CD414 1CD444, 1DD414 1DD444, 1DD413	129	200
10D423, 10D433	258	50

SURVEILLANCE REQUIREMENTS (continued)

d. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test.

f. At least once per 18 months, during shutdown, performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating. At this once per 18 months interval, this performance discharge test may be performed in lieu of the battery service test.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

PARAMETER	CATEGORY A: (*) LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: (*) LIMITS FOR EACH CONNECTED CELL	CATEGORY C: (#) ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte Level	≥Minimum level indication mark and <¼" above maximum level indication mark[(d)]	≥Minimum level indication mark and ≤¼" above maximum level indication mark[(d)]	Above top of plates and not overflowing
Float Voltage	≥2.13 volts	≥2.13 volts[(c)]	>2.07 volts
Specific Gravity[(a)]	≥1.200[(b)]	≥1.195 AND Average of all connected cells >1.205[(b)]	Not more than .020 below the average of all connected cells AND Average of all connected cells ≥1.195[(b)]

(*) With parameters of one or more cells in one or more batteries not within limits (i.e., Category A, Category B or Category A and B limits not met), the battery may be considered OPERABLE provided that:

1. Within 1 hour, pilot cell electrolyte levels and float voltages are verified to meet Category C Allowable Values, AND

2. Within 24 hours, and once per 7 days thereafter, all battery cell parameters meet Category C Allowable Values, AND

3. Within 31 days, all battery cell parameters are restored to within Category A and Category B limits of this Table.

(#) Any Category C parameter not within its Allowable Value indicates an inoperable battery.

(a) Corrected for electrolyte temperature and level.

(b) OR battery charging current is less than 2 amperes when on float charge.

(c) May be corrected for average electrolyte temperature.

(d) Electrolyte level may exceed ¼" above maximum level indication mark if an equalizing charge is in progress, or an equalizing charge has been completed within the previous 72 hours.

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D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following four channels of the D.C. electrical power sources, one of which shall be channel A or channel B, shall be OPERABLE with:

a. Channel A, consisting of:
   1. 125 volt battery 1AD411
   2. 125 volt full capacity charger# 1AD413 or 1AD414

b. Channel B, consisting of:
   1. 125 volt battery 1BD411
   2. 125 volt full capacity charger# 1BD413 or 1BD414.

c. Channel C, consisting of:
   1. 125 volt battery 1CD411
   2. 125 volt full capacity charger# 1CD413 or 1CD414
   3. 125 volt battery 1CD447
   4. 125 volt full capacity charger 1CD444

d. Channel D, consisting of:
   1. 125 volt battery 1DD411
   2. 125 volt full capacity charger# 1DD413 or 1DD414
   3. 125 volt battery 1DD447
   4. 125 volt full capacity charger 1DD444

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than two channels of the above required D.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

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*When handling recently irradiated fuel in the secondary containment.
#Only one full capacity charger per battery is required for the channel to be OPERABLE.

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system channels shall be energized:

a. A.C. power distribution:

1. Channel A, consisting of:

a) 4160 volt A.C. switchgear bus	10A401
b) 480 volt A.C. load centers	10B410 10B450
c) 480 volt A.C. MCCs	10B212 10B411 10B451 10B553
d) 208/120 volt A.C. distribution panels	10Y401(source:10B411) 10Y411(source:10B451) 10Y501(source:10B553)
e) 120 volt A.C. distribution panels	1AJ481 and inverter AD481 1YF401(source: 1AJ481) 1AJ482 and inverter AD482

2. Channel B, consisting of:

a) 4160 volt A.C. switchgear bus	10A402
b) 480 volt A.C. load centers	10B420 10B460
c) 480 volt A.C. MCCs	10B222 10B421 10B461 10B563
d) 208/120 volt A.C. distribution panels	10Y402(source:10B421) 10Y412(source:10B461) 10Y502(source:10B563)
e) 120 volt A.C. distribution panels	1BJ481 and inverter BD481 1YF402(source:1BJ481) 1BJ482 and inverter BD482

3. Channel C, consisting of:

a) 4160 volt A.C. switchgear bus	10A403
b) 480 volt A.C. load centers	10B430 10B470
c) 480 volt A.C. MCCs	10B232 10B431 10B471 10B573
d) 208/120 volt A.C. distribution panels	10Y403(source:10B431) 10Y413(source:10B471) 10Y503(source:10B573)

e) 120 volt A.C. distribution panels	1CJ481 and inverter CD481 1YF403(source:1CJ481) 1CJ482 and inverter CD482

4. Channel D, consisting of:

a) 4160 volt A.C. switchgear bus	10A404
b) 480 volt A.C. load centers	10B440 10B480
c) 480 volt A.C. MCCs	10B242 10B441 10B481 10B583
d) 208/120 volt A.C. distribution panels	10Y404(source:10B441) 10Y414(source:10B481) 10Y504(source:10B583)
e) 120 volt A.C. distribution panels	1DJ481 and inverter DD481 1YF404(source:1DJ481) 1DJ482 and inverter DD482

b. D.C. power distribution:

1. Channel A, consisting of:

a) 125 volt D.C. switchgear	10D410
b) 125 volt D.C. fuse box	1AD412
c) 125 volt D.C. distribution panel	1AD417
d) 250 volt D.C. switchgear	10D450
e) 250 volt D.C. fuse box	10D422
f) 250 volt D.C. MCC	10D251

2. Channel B, consisting of:

a) 125 volt D.C. switchgear	10D420
b) 125 volt D.C. fuse box	1BD412
c) 125 volt D.C. distribution panel	1BD417
d) 250 volt D.C. switchgear	10D460
e) 250 volt D.C. fuse boxes	10D432
f) 250 volt D.C. MCC	10D261

3. Channel C, consisting of:

a) 125 volt D.C. switchgear	10D430 10D436
b) 125 volt D.C. fuse box	1CD412 1CD448
c) 125 volt D.C. distribution panel	1CD417

4. Channel D, consisting of:

a) 125 volt D.C. switchgear	10D440 10D446
b) 125 volt D.C. fuse boxes	1DD412 1DD448
c) 125 volt D.C. distribution panel	1DD417

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above required A.C. distribution system channels not energized, re-energize the channel within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one of the above required 125 volt D.C. distribution system channels not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any one of the above required 250 volt D.C. distribution systems not energized, declare the associated HPCI or RCIC system inoperable and apply the appropriate ACTION required by the applicable Specifications.

d. With one or both inverters in one channel inoperable, energize the associated 120 volt A.C. distribution panel(s) within 8 hours, and restore the inverter(s) to OPERABLE status within 24 hours; or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system channels shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker/switch alignment and voltage on the busses/MCCs/panels.

DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, 2 of the 4 channels, one of which shall be channel A or channel B, of the power distribution system shall be energized with:

a. A.C. power distribution:

1. Channel A, consisting of:

a)	4160 volt A.C. switchgear bus	10A401
b)	480 volt A.C. load centers	10B410 10B450
c)	480 volt A.C. MCCs	10B212 10B411 10B451 10B553
d)	208/120 volt A.C. distribution panels	10Y401(source:10B411) 10Y411(source:10B451) 10Y501(source:10B553)
e)	120 volt A.C. distribution panels	1AJ481 1YF401(source:1AJ481) 1AJ482

2. Channel B, consisting of:

a)	4160 volt A.C. switchgear bus	10A402
b)	480 volt A.C. load centers	10B420 10B460
c)	480 volt A.C. MCCs	10B222 10B421 10B461 10B563
d)	208/120 volt A.C. distribution panels	10Y402(source:10B421) 10Y412(source:10B461) 10Y502(source:10B563)
e)	120 volt A.C. distribution panels	1BJ481 1YF402(source:1BJ481) 1BJ482

3. Channel C, consisting of:

a)	4160 volt A.C. switchgear bus	10A403
b)	480 volt A.C. load centers	10B430 10B470
c)	480 volt A.C. MCCs	10B232 10B431 10B471 10B573
d)	208/120 volt A.C. distribution panels	10Y403(source:10B431) 10Y413(source:10B471) 10Y503(source:10B573)

	e)	120 volt A.C. distribution panels	1CJ481 1YF403(source:1CJ481) 1CJ482
4.	Channel D, consisting of:		
	a)	4160 volt A.C. switchgear bus	10A404
	b)	480 volt A.C. load centers	10B440 10B480
	c)	480 volt A.C. MCCs	10B242 10B441 10B481 10B583
	d)	208/120 volt A.C. distribution panels	10Y404(source:10B441) 10Y414(source:10B481) 10Y504(source:10B583)
	e)	120 volt A.C. distribution panels	1DJ481 1YF404(source:1DJ481) 1DJ482

b. D.C. power distribution:

1.	Channel A, consisting of:		
	a)	125 volt D.C. switchgear	10D410
	b)	125 volt D.C. fuse box	1AD412
	c)	125 volt D.C. distribution panel	1AD417
2.	Channel B, consisting of:		
	a)	125 volt D.C. switchgear	10D420
	b)	125 volt D.C. fuse box	1BD412
	c)	125 volt D.C. distribution panel	1BD417
3.	Channel C, consisting of:		
	a)	125 volt D.C. switchgear	10D430 10D436
	b)	125 volt D.C. fuse boxes	1CD412 1CD448
	c)	125 volt D.C. distribution panel	1CD417
4.	Channel D, consisting of:		
	a)	125 volt D.C. switchgear	10D440 10D446
	b)	125 volt D.C. fuse box	1DD412 1DD448
	c)	125 volt D.C. distribution panel	1DD417

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than two channels of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With less than two channels of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system channels shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker/switch alignment and voltage on the busses/MCCs/panels.

*When handling recently irradiated fuel in the secondary containment.

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment penetration conductor over current protective devices shown in Table 3.8.4.1-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system, and

   1. For 4.16 kV circuit breakers, de-energize the 4.16 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

   2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by disconnecting* the breaker within 72 hours and verify the inoperable breaker(s) to be disconnected at least once per 7 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program:

   1. By verifying that each of the medium voltage 4.16 kV circuit breakers are OPERABLE by performing:

      a) A CHANNEL CALIBRATION of the associated protective relays, and

      b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

* After being disconnected, these breakers shall be maintained disconnected under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value between 150% and 300% of the pickup of the long time delay trip element and verifying that the circuit breaker operates within the time delay bandwidth for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of 120% the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. In accordance with the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8.4.1-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

1. 4160-VOLT CIRCUIT BREAKERS

CIRCUIT BREAKER NO.	LOCATION	SYSTEMS OR EQUIPMENT POWERED
1AN205	1AN205	Reactor Recirculation Pump 1AP201
1BN205	1BN205	Reactor Recirculation Pump 1BP201
1CN205	1CN205	Reactor Recirculation Pump 1AP201
1DN205	1DN205	Reactor Recirculation Pump 1BP201

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS

Primary and backup breakers have the same device numbers and are located in the same Motor Control Center cubicle.

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-411065	10B411	IM TM	RHR Head Spray Valve 1BC-HV-F022
52-451061	10B451	IM TM	RHR Shutdown Cooling Suction Inboard Valve 1BC-HV-F009
52-212021	10B212	IM TM	RWCU Suction Isolation Inboard Valve 1BG-HV-F001
52-212101	10B212	IM TM	PCIGS Drywell Supply Header A Isolation Valve 1KL-HV-5152A
52-212181	10B212	IM TM	Main Steam Line Drain Inboard Valve 1AB-HV-F016
52-212183	10B212	IM TM	PCIGS Drywell Suction Inboard Valve 1KL-HV-5148
52-232061	10B232	IM TM	Drywell Supply Header A Isolation Valve 1KL-HV-5124A
52-232103	10B232	IM TM	Drywell Equip. Drain Sump Isolation Valve 1HB-HV-F019
52-232104	10B232	IM TM	HPCI Warmup Bypass Line Isolation Valve 1FD-HV-F100
52-232181	10B232	IM TM	Chilled Water Loop A Supply Isolation Valve 1GB-HV-9531B1
52-232182	10B232	IM TM	Chilled Water Loop A Return Isolation Valve 1GB-HV-9531B2
52-232183	10B232	IM TM	Chilled Water Loop B Supply Isolation Valve 1GB-HV-9531B3

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS (Continued)

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-232193	10B232	IM TM	Chilled Water Loop B Return Isolation Valve 1GB-HV-9531B4
52-232203	10B232	IM TM	HPCI Turbine Steam Supply Isolation Valve 1FD-HV-F002
52-242021	10B242	IM TM	Drywell Floor Drain Sump Isolation Valve 1HB-HV-F003
52-242061	10B242	IM TM	Drywell Supply Header B Isolation Valve 1KL-HV-5124B
52-242101	10B242	IM TM	PCIGS Drywell Supply Header B Isolation Valve 1KL-HV-5152B
52-242102	10B242	IM TM	RCIC Turbine Steam Supply Isolation Valve 1FC-HV-F007
52-242103	10B242	IM TM	RCIC Warmup Bypass Line Isolation Valve 1FC-HV-F076
52-242172	10B242	IM TM	Reactor Recirc Pumps Cooling Supply Isolation 1ED-HV-2554
52-242173	10B242	IM TM	Reactor Recirc Pumps Cooling Return Isolation 1ED-HV-2556
52-252021	10B252	IM TM	Drywell Cooler A Fan 1A1V212
52-252022	10B252	IM TM	Drywell Cooler B Fan 1B1V212
52-252031	10B252	IM TM	Drywell Cooler C Fan 1C1V212
52-252032	10B252	IM TM	Drywell Cooler D Fan 1D1V212
52-252041	10B252	IM TM	Drywell Cooler E Fan 1E1V212
52-252042	10B252	IM TM	Drywell Cooler F Fan 1F1V212
52-252051	10B252	IM TM	Drywell Cooler G Fan 1G1V212
52-252052	10B252	IM TM	Drywell Cooler H Fan 1H1V212

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS (Continued)

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-252063	10B252	IM TM	Drywell Equip Drain Sump Pump 1AP267
52-252064	10B252	IM TM	Drywell Floor Drain Sump Pump 1CP267
52-252073	10B252	IM TM	Feedwater Inlet A Shutoff 1AE-HV-F011A
52-262021	10B262	IM TM	Drywell Cooler A Fan 1A2V212
52-262022	10B262	IM TM	Drywell Cooler B Fan 1B2V212
52-262031	10B262	IM TM	Drywell Cooler C Fan 1C2V212
52-262032	10B262	IM TM	Drywell Cooler D Fan 1D2V212
52-262041	10B262	IM TM	Drywell Cooler E Fan 1E2V212
52-262042	10B262	IM TM	Drywell Cooler F Fan 1F2V212
52-262051	10B262	IM TM	Drywell Cooler G Fan 1G2V212
52-262052	10B262	IM TM	Drywell Cooler H Fan 1H2V212
52-262063	10B262	IM TM	Drywell Equip Drain Sump Pump 1BP267
52-262064	10B262	IM TM	Drywell Floor Drain Sump Pump 1DP267
52-253012*	10B253	IM TM	Recirc Pump Motor Hoist 1AH201 Disconnect Switch 1AS204
52-253021	10B253	IM TM	Recirc Pump 1BP201 Suction Valve 1BB-HV-F023B
52-253031	10B253	IM TM	Recirc Pump 1BP201 Discharge Valve 1BB-HV-F031B
52-253053	10B253	IM TM	Reactor Vessel Head Vent Inboard Isolation 1BB-HV-F001

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS (Continued)

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-253064	10B253	IM TM	Reactor Vessel Head Vent to Steam Line 1BB-HV-F005
52-263011	10B263	IM TM	Reactor Vessel Head Vent Outboard Isolation 1BB-HV-F002
52-263012*	10B263	IM TM	Recirc Pump Motor Hoist 1BH201 Disconnect Switch 1BS204
52-263042*	10B263	IM TM	Main Steam Relief Valve Hoist 10H202 Disconnect Switch 10S207
52-263054	10B263	IM TM	RWCU Suction from Recirc Loop A 1BG-HV-F100
52-263081	10B263	IM TM	RWCU Suction from RPV Drain Valve 1BG-HV-F101
52-263082	10B263	IM TM	RWCU Suction Valve 1BG-HV-F102
52-263083	10B263	IM TM	RWCU Suction from Recirc Loop B Valve 1BG-HV-F106
52-264053	10B264	IM TM	Recirc Pump A Discharge Valve 1BB-HV-F031A
52-264062	10B264	IM TM	Feedwater Inlet B Shutoff Valve 1AE-HV-F011B
52-264071	10B264	IM TM	Reactor Recirc Pump 1AP201 Space Heater 1AS220
52-264072	10B264	IM TM	Reactor Recirc Pump 1BP201 Space Heater 1BS220
52-264033	10B264	IM TM	Recirc Pump A Suction Valve 1BB-HV-F023A

* These breakers shall be administratively maintained open in OPERATIONAL CONDITIONS 1, 2 and 3 and are not required to be tested.

ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection bypass circuit of each motor operated valve (MOV) required to have thermal overload protection shall be OPERABLE.

APPLICABILITY: Whenever the MOV is required to be OPERABLE.

ACTION:

With the thermal overload protection bypass circuit for one or more of the above required MOVs inoperable, restore the inoperable thermal overload protection bypass circuit(s) to OPERABLE status within 8 hours or declare the affected MOV(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection bypass circuit for each of the above required MOVs shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by the performance of a CHANNEL FUNCTIONAL TEST for:

    1. Those thermal overload protection devices which are normally in force during plant operation and bypassed only under accident conditions.

    2. A representative sample of at least 25% of those thermal overload protection devices which are bypassed continuously and temporarily placed in force only when the MOVs are undergoing periodic or maintenance testing.

    3. A representative sample of at least 25% of those thermal overload protection devices which are in force during normal manual (momentary push button contact) MOV operation and bypassed during remote manual (push button held depressed) MOV operation.

b. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection for the above required MOVs which are continuously bypassed and temporarily placed in force only when the MOV is undergoing periodic or maintenance testing shall be verified to be continuously bypassed following such testing.

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MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (NOT BYPASSED)

LIMITING CONDITION FOR OPERATION

3.8.4.3 The thermal overload protection of each motor operated valve (MOV) shown in Table 3.8.4.3-1 shall be OPERABLE.

APPLICABILITY: Whenever the MOV is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required MOVs inoperable, restore the inoperable thermal overload(s) to OPERABLE status within 8 hours or declare the affected MOV(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for each of the above required MOVs shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION.

TABLE 3.8.4.3-1

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (NOT BYPASSED)

VALVE NUMBER	SYSTEM(S) AFFECTED
1BC-HV-F003A	Residual Heat Removal
1BC-HV-F003B	Residual Heat Removal
1GS-HV-5741A	Containment Atmosphere Control
1GS-HV-5741B	Containment Atmosphere Control
1KC-HV-3408M	Fire Protection

REACTOR PROTECTION SYSTEM ELECTRICAL POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.4 Two RPS electric power monitoring channels for each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS electric power monitoring channels shall be determined OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.

b. In accordance with the Surveillance Frequency Control Program by demonstrating the OPERABILITY of over-voltage, under-voltage, and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   1. Over-voltage ≤ 132 VAC, (Bus A), 132 VAC (Bus B)

   2. Under-voltage ≥ 108 VAC, (Bus A), 108 VAC (Bus B)

   3. Under-frequency ≥ 57 Hz. (Bus A and Bus B)

CLASS 1E ISOLATION BREAKER OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.5 All Class 1E isolation breaker (tripped by a LOCA signal) overcurrent protective devices shown in Table 3.8.4.5-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the overcurrent protective devices shown in Table 3.8.4.5-1 inoperable, declare the affected isolation breaker inoperable and remove the inoperable circuit breaker(s) from service within 72 hours and verify the inoperable breaker(s) to be disconnected at least once per 7 days thereafter.

SURVEILLANCE REQUIREMENTS

4.8.4.5 Each of the Class 1E isolation breaker overcurrent protective devices shown in Table 3.8.4.5-1 shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program:

   By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value between 150% and 300% of the pickup of the long time delay trip element and a value between 150% and 250% of the pickup of the short time delay, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of 120% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. For circuit breakers equipped with solid state trip devices, the functional testing may be performed with use of portable instruments designed to verify the time-current characteristics and pickup calibration of the trip elements. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. In accordance with the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8.4.5-1

CLASS 1E ISOLATION BREAKER

OVERCURRENT PROTECTIVE DEVICES
(BREAKER TRIPPED BY A LOCA SIGNAL)

480 VAC POWER CIRCUIT BREAKERS

1.

Class 1E Circuit Breaker No.	Class 1E Bus	Non-Class 1E Load Description
52-41011	10B410	Reactor Auxiliaries Cooling System Pump 1AP209
52-41014	10B410	Radwaste and Service Area MCC 10B313
52-41024	10B410	Reactor Building Supply Air Handling Unit 1BVH300
52-42011	10B420	Reactor Auxiliaries Cooling System Pump 1BP209
52-42014	10B420	Radwaste and Service Area MCC 10B323
52-42024	10B420	Reactor Building Exhaust Fan 1BV301
52-43024	10B430	Reactor Building Supply Air Handling Unit 1CVH300
52-43014	10B430	Control Rod Drive Pump 1AP207
52-44014	10B440	Control Rod Drive Pump 1BP207
52-44024	10B440	Reactor Building Supply Air Handling Unit 1AVH300
52-44034	10B440	Radwaste Area Supply Fan 0BV316
52-45011	10B450	Reactor Area MCC 10B252
52-45014	10B450	Radwaste Area Exhaust Fan OAV305
52-45024	10B450	Emergency Instrument Air Compressor 10K100

TABLE 3.8.4.5-1 (Continued)

480 VAC POWER CIRCUIT BREAKERS

1. (Continued)

Class 1E Circuit Breaker No.	Class 1E Bus	Non-Class 1E Load Description
52-45034	10B450	Reactor Building Exhaust Fan 1CV301
52-46011	10B460	Reactor Area MCC 10B262
52-46014	10B460	Radwaste Area Exhaust Fan 0BV305
52-47011	10B470	Reactor Area MCC 10B272
52-47014	10B470	Radwaste Area Exhaust Fan 0CV305
52-47024	10B470	Radwaste Area Supply Fan 0AV316
52-47031	10B470	Technical Support Center MCC 00B474
52-48011	10B480	Reactor Area MCC 10B282
52-48024	10B480	Reactor Building Exhaust Fan 1AV301

480 VAC MOLDED CIRCUIT BREAKERS

1.

Class 1E Circuit Breaker No.	Class 1E Bus	Non-Class 1E Load Description
52-441043	10B441	NSSS Computer Inverter 10D485
52-451023	10B451	Public Address System Inverter 10D496
52-471023	10B471	Security System Inverter 0AD495

POWER RANGE NEUTRON MONITORING SYSTEM ELECTRICAL POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.6 The power range neutron monitoring system (NMS) electric power monitoring channels for each inservice power range NMS power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one power range NMS electric power monitoring channel for an inservice power range NMS power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or deenergize the associated power range NMS power supply feeder circuit.

b. With both power range NMS electric power monitoring channels for an inservice power range NMS power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or deenergize the associated power range NMS power supply feeder circuit.

SURVEILLANCE REQUIREMENTS

4.8.4.6 The above specified power range NMS electric power monitoring channels shall be determined OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.

b. In accordance with the Surveillance Frequency Control Program by demonstrating the OPERABILITY of over-voltage, under-voltage, and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

1. Over-voltage ≤ 132 VAC (BUS A), 132 VAC (BUS B)

2. Under-voltage ≥ 108 VAC (BUS A), 108 VAC (BUS B)

3. Under-frequency ≥ 57 Hz. -0, +2%

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform main hoist fuel-loaded.
4. Service platform hoist fuel-loaded.

APPLICABILITY: OPERATIONAL CONDITION 5* #.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

* See Special Test Exceptions 3.10.1 and 3.10.3.

The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours prior to:

1. Beginning CORE ALTERATIONS, and

2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. In accordance with the Surveillance Frequency Control Program.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

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* The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:##

a. Annunciation and continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn.#

d. During a SPIRAL UNLOAD, the count rate may drop below 3 cps when the number of assemblies remaining in the core drops to sixteen or less.

e. During a SPIRAL RELOAD, up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps. Until these assemblies have been loaded, the 3 cps count rate is not required.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and fully insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. In accordance with the Surveillance Frequency Control Program:

1. Performance of a CHANNEL CHECK,

* The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

\# Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

\## Three SRM channels shall be OPERABLE for critical shutdown margin demonstrations. An SRM detector may be retracted provided a channel indication of at least 100 cps is maintained.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the detectors are inserted to the normal operating level, and

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

c. Verifying that the channel count rate is at least 3 cps.

1. Prior to control rod withdrawal,

2. Prior to and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS***, and

3. In accordance with the Surveillance Frequency Control Program***.

d. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, verifying that the RPS circuitry "shorting links" have been removed, within 8 hours prior to and in accordance with the Surveillance Frequency Control Program during the time any control rod is withdrawn.**

** Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

*** Except as noted in Specifications 3.9.2.d and 3.9.2.e.

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified:

a. Within 2 hours prior to:

   1. The start of CORE ALTERATIONS.

   2. The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.

b. In accordance with the Surveillance Frequency Control Program.

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* Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** See Special Test Exception 3.10.3.

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3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet 2 inches of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth in accordance with the Surveillance Frequency Control Program.

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

f. All fuel loading operations shall be suspended.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and in accordance with the Surveillance Frequency Control Program thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

f. All fuel loading operations are suspended.

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors SRM are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations shall be suspended.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and in accordance with the Surveillance Frequency Control Program thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations are suspended.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet 2 inches above the top of the reactor pressure vessel flange and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

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* The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

** Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each loop consisting of:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet 2 inches above the top of the reactor pressure vessel flange and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

a. With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the OPERABILITY of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

* The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

** Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program during low power PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.2 ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod worth minimizer (RWM) per Specification 3.1.4.1 may be suspended for the following tests provided that control rod movement prescribed for this testing is verified by a second licensed operator or other technically qualified member of the unit technical staff present at the reactor console:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 when THERMAL POWER is less than or equal to 8.6% of RATED THERMAL POWER.

ACTION:

With the requirements of the above specification not satisfied, verify that the RWM is OPERABLE per Specifications 3.1.4.1.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed by the RWM are bypassed, verify:

a. That movement of the control rods from 75% ROD DENSITY to the RWM low power setpoint is limited to the approved control rod withdrawal sequence during scram and friction tests.

b. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3.

c. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

SPECIAL TEST EXCEPTIONS

3/4 10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation with matched pump speed may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

The material originally contained in this Technical Specification was deleted with the issuance of Amendment No.35. However, to maintain the historical reference to this specification, this section has been intentionally left blank.

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program during training startups.

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

LIMITING CONDITION FOR OPERATION

3/4.10.7 The material originally contained in Section 3/4.10.7 was deleted with the issuance of Amendment No. 14. However, to maintain the historical reference to this section, Section 3/4.10.7 is intentionally left blank.

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

LIMITING CONDITION FOR OPERATION

3.10.8 When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to 212°F, and operation considered not to be in OPERATIONAL CONDITION 3, to allow performance of an inservice leak or hydrostatic test provided the following OPERATIONAL CONDITION 3 LCO's are met:

a. 3.3.2, "ISOLATION ACTUATION INSTRUMENTATION", Functions 2.a, 2.c, 2.d and 2.e of Table 3.3.2-1;

b. 3.6.5.1, "SECONDARY CONTAINMENT INTEGRITY";

c. 3.6.5.2, "SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS"; and

d. 3.6.5.3, "FILTRATION, RECIRCULATION AND VENTILATION SYSTEM."

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature > 200°F.

ACTION:

With the requirements of the above specification not satisfied, immediately enter the applicable condition of the affected specification or immediately suspend activities that could increase the average reactor coolant temperature or pressure and reduce the average reactor coolant temperature to ≤ 200°F within 24 hours.

SURVEILLANCE REQUIREMENTS

4.10.8 Verify applicable OPERATIONAL CONDITION 3 surveillances for specifications listed in 3.10.8 are met.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

3.11.1.1 Deleted

3.11.1.2 Deleted

3.11.1.3 Deleted

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

3.11.2.1 Deleted

3.11.2.2 Deleted

3.11.2.3 Deleted

3.11.2.4 Deleted

3.11.2.5 Deleted

3.11.2.6 Deleted

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MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.7 The radioactivity rate of noble gases measured at the recombiner after-condenser discharge shall be limited to less than or equal to 3.30 E+5 microcuries/sec after 30 minute decay.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3*.

ACTION:

With the radioactivity rate of noble gases at the recombiner after-condenser discharge exceeding 3.30 E+5 microcuries/sec after 30 minute decay, restore the radioactivity rate to within its limit within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases at the recombiner after-condenser discharge shall be continuously monitored in accordance with Specification 3.3.7.1.

4.11.2.7.2 The radioactivity rate of noble gases from the recombiner after-condenser discharge shall be determined to be within the limits of Specification 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken near the discharge of the main condenser air ejector:

a. In accordance with the Surveillance Frequency Control Program.

b. Within 4 hours following an increase, as indicated by the Offgas Pretreatment Radiation Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady-state fission gas release from the primary coolant.

c. The provisions of Specification 4.0.4 are not applicable.

* When the main condenser air ejector is in operation.

3.11.2.8 Deleted

3.11.3 Deleted

3.11.4 Deleted

PAGES 3/4 11-19 THROUGH 3/4 11-20 HAVE BEEN DELETED

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 Deleted

3/4.12.2 Deleted

3/4.12.3 Deleted

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SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA AND MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1. Information regarding radioactive gaseous and liquid effluents which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1. The circle with the five mile radius is the low population zone.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel structure composed of a spherical lower portion, a cylindrical middle portion, and a hemispherical top head which form a drywell. The drywell is attached to the suppression chamber through a series of downcomer vents. The suppression chamber is a steel pressure vessel in the shape of a torus. The drywell has a nominal free air volume of 169,000 cubic feet. The suppression chamber has an air volume of 137,000 cubic feet and a water region as described in Technical Specification Bases 3/4.6.2, Depressurization Systems.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

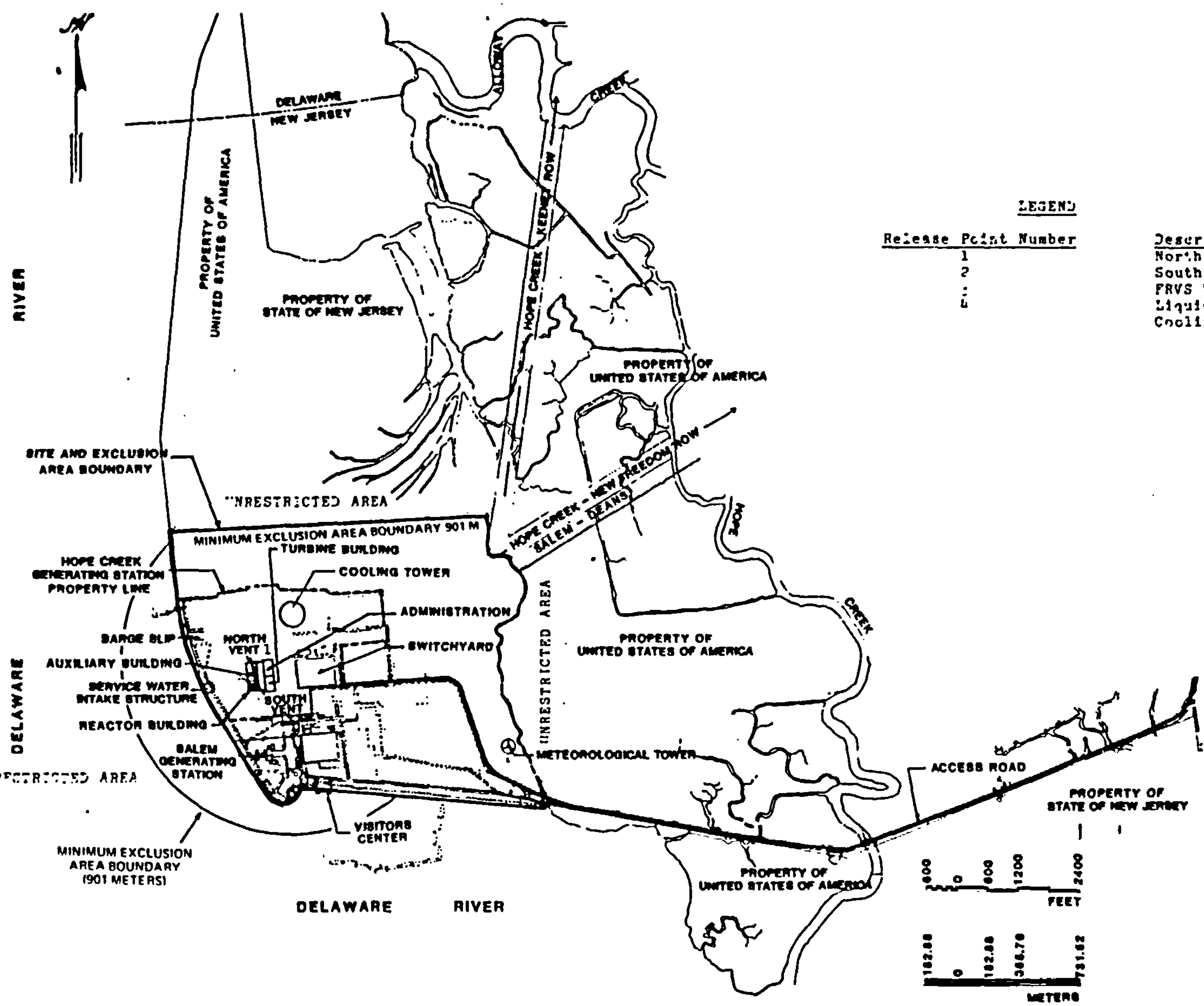
a. Maximum internal pressure 62 psig.

b. Maximum internal temperature: drywell 340°F.
suppression pool 310°F.

c. Maximum external differential pressure 3 psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building, and a portion of the main steam tunnel and has a free volume of 4,000,000 cubic feet.



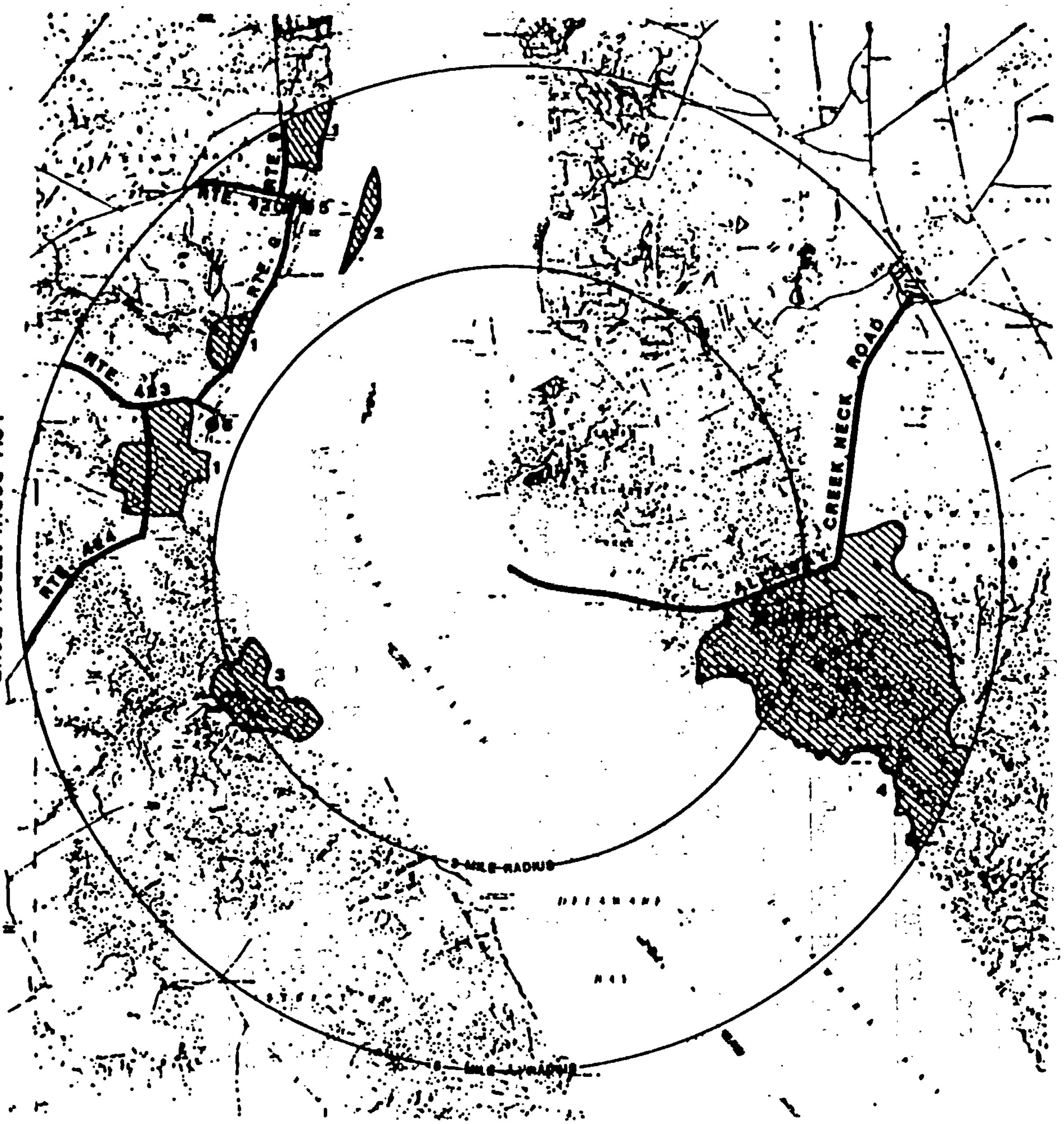
LEGEND

Release Point Number	Description
1	North Plant Vent
2	South Plant Vent
3	FRVS Vent
4	Liquid Radwaste Discharge & Cooling Tower Blowdown Line

FIGURE 5.1.1-1

EXCLUSION AREA AND UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

LOW POPULATION ZONE
1982



TRANSPORTATION: HIGHWAY

DELAWARE 9
420
423
424
NEW JERSEY ALLOWAY CREEK NECK ROAD

RIVER

DELAWARE RIVER

WILDLIFE AREAS

1. AUGUSTINE CREEK WILDLIFE AREA
2. REEDY ISLAND WILDLIFE REFUGE
3. APPOQUINIMINK WILDLIFE AREA
4. MAD HORSE CREEK WILDLIFE MANAGEMENT AREA

BEACH AREAS

5. AUGUSTINE BEACH
6. BAY VIEW BEACH

SOURCE: NEW CASTLE COUNTY PLANNING BOARD, THE RED LION PLANNING DISTRICT PLAN, 1995, SEPTEMBER 1973

NEW CASTLE COUNTY PLANNING BOARD, THE MIDDLETOWN ODESSA-TOWNSEND PLANNING DISTRICT PLAN, 1995, SEPTEMBER 1973

R. CHARTAWICH, DEPT. OF PLANNING NEW CASTLE CO PLANNING BOARD, MAY 1982

C. WARREN, DEPT. OF LAND USE & POPULATION, SALE COUNTY PLANNING BOARD, APRIL 1982

U.S. DEPT. OF AGRICULTURE, SOUTH JERSEY RESOURC CONSERVATION & DEVELOPMENT AREA PLAN, APRIL 1

HOPE CREEK

FIGURE 5.1.2-1

LOW POPULATION ZONE

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material and water rods. Limited substitutions of zirconium 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 NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

A maximum of twelve GE14i Isotope Test Assemblies may be placed in non-limiting core regions, beginning with Reload 16 Cycle 17 core reload, with the purpose of obtaining surveillance data to verify that the GE14i cobalt Isotope Test Assemblies perform satisfactorily in service (prior to evaluating a future license amendment for use of these design features on a production basis). Each GE14i assembly contains a small number of Zircaloy-2 clad isotope rods containing Cobalt-59. Cobalt-59 targets will transition into Cobalt-60 isotope targets during cycle irradiation of the assemblies. Details of the GE14i assemblies are contained in GE-Hitachi report NEDC-33529P, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," Revision 0, dated December 2009.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The absorber material has a nominal absorber length of 143 inches.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump.
   2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

5.4 REACTOR COOLANT SYSTEM (continued)

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 21,970 cubic feet at a nominal steam dome saturation temperature of 547°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal 6.308 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 199' 4".

CAPACITY

5.6.3 The spent fuel storage pool shall be limited to a storage capacity of no more than 4006 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 546°F to 70°F
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	130 hydrostatic pressure and leak tests	Pressurized to ≥ 930 and ≤1250 psig

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Senior Nuclear Shift Supervisor, or during his absence from the control room, a designated individual shall be responsible for the control room command function. A management directive to this effect, signed by the senior corporate nuclear officer shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 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 positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined from the highest management levels through-intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Hope Creek Generating Station Updated Final Safety Analysis Report and updated in accordance with 10 CFR 50.71(e).

b. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The senior corporate nuclear 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.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

6.2.2 The unit organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

UNIT STAFF (Continued)

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator shall be in the control room;

c. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

FIGURE 6.2.1-1
DELETED

FIGURE 6.2.2-1
DELETED

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SNSS*	1	1
NSS*	1	None
NCO	2	1
EO	2	1
STA	1	None
RPT	1	1

TABLE NOTATION

SNSS - Senior Nuclear Shift Supervisor with a Senior Reactor Operator license on the Unit
NSS - Nuclear Shift Supervisor with a Senior Reactor Operator license on the Unit
NCO - Nuclear Control Operator with a Reactor Operator license on the Unit
EO - Equipment Operator
STA - Shift Technical Advisor
RPT - Radiation Protection Technician

Except for the Senior Nuclear Shift Supervisor, the shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 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 of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Senior Nuclear Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual with a valid Senior Reactor Operator license shall be designated to assume the control room command function. During any absence of the Senior Nuclear Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Reactor Operator license or Operator license shall be designated to assume the control room command function.

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*In cases where an individual has a Senior Reactor Operator's license on the unit, is a qualified STA, and has a Professional Engineers License by virtue of successful completion of the Professional Engineers examination or a bachelor's degree in a scientific, engineering, or engineering technology discipline from an accredited institution, the individual can serve in a dual role capacity as either the SNSS/STA or NSS/STA. (Note: For those individuals with a bachelor's degree in a scientific or engineering technology discipline, course work must have included physical, mathematical, or engineering science.) Otherwise, there shall be a qualified STA as well as a SNSS and NSS on-shift.

6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall provide advisory technical support to the Senior Nuclear Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1981 for comparable positions, except for the individual designated as the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, the individual designated as the operations manager who shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1981 except as modified by Specification 6.3.3, and the licensed operators who shall comply with the requirements of 10 CFR Part 55.

6.3.2 The Operations Manager or Assistant Operations Manager shall hold a senior reactor operator license. The Senior Nuclear Shift Supervisors, and Nuclear Shift Supervisors, shall hold a senior reactor operator license. The Nuclear Control Operators shall hold a reactor operator license.

6.3.3 The Operations Manager shall meet one of the following:

(1) Hold a senior reactor operator license, or

(2) Have held a senior reactor operator license for this or a similar unit (BWR), or

(3) Have been certified at an appropriate simulator for equivalent senior operator knowledge.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall: 1) be maintained under the direction of the Director - Nuclear Training, 2) shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS 3.1-1981 for all affected positions except licensed operators, and 3) comply with the requirements of 10 CFR Part 55 for licensed operators.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Fire Protection Manager and shall meet or exceed the requirements of the SRP (NUREG-0800) Section 13.2.2.II.6, 10 CFR 50 Appendix R and Branch Technical Position CMEB 9.5.1, Section C.3.d.

6.5 REVIEW AND AUDIT (THIS SECTION DELETED)

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6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified pursuant to the requirements of Section 50.72 to 10 CFR Part 50 and a report submittal pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Station Operations Review Comittee (SORC), and the results of this review shall be submitted to the Nuclear Review Board and the senior corporate nuclear officer.

6.7. SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The senior corporate nuclear officer and the senior management position with responsibility for independent nuclear safety assessment activities and quality program oversight shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the senior management position with responsibility for independent nuclear safety assessment activities and quality program oversight and the senior corporate nuclear officer within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto.

c. Refueling operations.

d. Surveillance and test activities of safety-related equipment.

e. Security Plan implementation.

f. Emergency Plan implementation.

g. Fire Protection Program implementation.

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environment monitoring.

6.8.2 Each procedure and administrative policy of 6.8.1 above, except 6.8.1.e and 6.8.1.f, and changes thereto, shall be reviewed and approved in accordance with requirements in Updated Final Safety Analysis Report (UFSAR) section 17.2 for SORC or for Technical Review and Control, as appropriate, prior to implementation and reviewed periodically as set forth in administrative procedures. Procedures of 6.8.1.e and 6.8.1.f shall be reviewed and approved in accordance with the Facility's Security and Emergency Plans or requirements in Updated Final Safety Analysis Report (UFSAR) section 17.2 for Technical Review and Control, as appropriate, prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 On-the-Spot changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

c. The change is documented and receives the same level of review and approval as the original procedure within 14 days of implementation.

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the HPCI, CS, RHR, RCIC, Containment Hydrogen Recombiner, H2/02 analyzer, Post-Accident Sampling, Control Rod Drive Hydraulic (Scram Discharge portion) systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. A service pressure leak test for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

c. Deleted

d. Explosive Gas Monitoring

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System. The program shall include the limit for hydrogen concentration in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limit is maintained. This limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion).

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Explosive Gas Monitoring Program surveillance frequencies.

e. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. an API gravity or absolute specific gravity within limits for ASTM 2D fuel oil,

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. bulk water and sediment within limits for ASTM 2D fuel oil;

b. Other properties for new ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the stored fuel oil is $\leq$ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, modified as follows: The 0.8 micron membrane filters specified in ASTM D-2276 may be replaced with membrane filters up to 3.0 microns.

6.8.4.f Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, 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 the following exception:

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after April 12, 1994 shall be performed no later than April 12, 2009.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 50.6 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 0.5% of primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary containment leakage rate acceptance criterion is less than or equal to 1.0 La. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.6 La for Type B and Type C tests and less than or equal to 0.75 La for Type A tests;

b. Air lock testing acceptance criteria are:
   1) Overall air lock leakage rate is less than or equal to 0.05 La when tested at greater than or equal to Pa,
   2) Door seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to greater than or equal to 10.0 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

6.8.4.g. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBER(S) OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

PROCEDURES AND PROGRAMS (Continued)

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 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 from radioactive materials in liquid effluents released from the unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected 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,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

PROCEDURES AND PROGRAMS (Continued)

6.8.4.g. Radioactive Effluent Controls Program

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

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 greater than 8 days in gaseous effluents released from the unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on venting and purging of the containment through the Reactor Building Ventilation System, Hardened Torus Vent, or the FRVS to maintain releases as low as reasonably achievable, and

11) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

h. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluents monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

PROCEDURES AND PROGRAMS (Continued)

6.8.4.i INSERVICE TESTING PROGRAM

This Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. 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

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies and to 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 Specification 4.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

6.8.4.j Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the USNRC Administrator, Region 1, unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.

6.9.1.5 Reports required on an annual basis shall include:

a. Deleted

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

b. Documentation of all challenges to main steamline safety/relief valves.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 The Annual Radiological Environmental Operating report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis 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 (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.7 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid wastes released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

6.9.1.8 Deleted

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the PSEG Nuclear LLC generated CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following Technical Specifications:

3/4.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE
3/4.2.3	MINIMUM CRITICAL POWER RATIO
3/4.2.4	LINEAR HEAT GENERATION RATE
3/4.3.11	OSCILLATION POWER RANGE MONITOR (OPRM)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC as applicable in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

2. CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology"

3. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, August 1996

The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number title, revision, date, and any supplements).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 DELETED

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

RECORD RETENTION (Continued)

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections.

i. Records of quality assurance activities required by the Quality Assurance Program.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of SORC meetings and activities of the Nuclear Review Board (and activities of its predecessor, the Offsite Safety Review (OSR) staff).

l. DELETED

m. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analyses at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATIONAL MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 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:

6.12.1 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

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

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.

6.12.2 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

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 supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

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 or group entering such an area shall possess:

1. 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

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 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 and control every individual in the area.

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 displays 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.

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3.n. This documentation shall contain:

   (1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the changes(s) and

   (2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the SORC and the approval of the Plant General Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3.n. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the changes(s) and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the SORC and the approval of the Plant General Manager.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to 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 (e.g., month/year) the change was implemented.

6.15 TECHNICAL SPECIFICATION (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.

b. PSEG may make changes to the 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. Proposed changes to the Bases that require either condition of Specification 6.15.b above shall be reviewed and approved by the NRC prior to implementation.

d. 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).

e. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

6.16 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System, CRE 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 CRE 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 CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE 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 CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

6.16 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM (Continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the Control Room Emergency Filtration System, operating at the flow rate required by Surveillance Requirement 4.7.2.1.c.1, at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 36 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. 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 CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of Specification 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. NPF-57

HOPE CREEK GENERATING STATION

PSEG NUCLEAR LLC

DOCKET NO. 50-354

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

HOPE CREEK GENERATING STATION
ENVIRONMENTAL PROTECTION PLAN
(NONRADIOLOGICAL)

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1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of nonradiological environmental values during operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement - Operating Licensing Stage (FES-OL) and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters are regulated by way of the licensee's NPDES permit.

2.0 Environmental Protection Issues

In the FES-OL dated December 1984, the staff considered the environmental impacts associated with the operation of the Hope Creek Generating Station. Certain environmental issues were identified which required study or license conditions to resolve environmental concerns and to assure adequate protection of the environment.

2.1 Aquatic/Water Quality Issues

Consumptive surface water use by Hope Creek during periods of river flow below 85 m^3/s (3,000 ft^3/s), as measured at Trenton, New Jersey, is to be compensated for under a ruling of the Delaware River Basin Commission (DRBC). The applicant is participating in the development of a supplementary reservoir for this purpose. (FES Section 4.2.3.2 and 5.3.1.1). The NRC will defer to the DRBC for any further actions regarding flow compensation.

2.2 Terrestrial Issues

The primary potential effect of station operation on terrestrial resources derives from cooling tower drift. Significant impacts on terrestrial resources will likely not occur if the cooling tower functions properly and is adequately maintained. To ensure proper cooling tower operation, the need to measure drift rates and deposition on native vegetation was identified by the staff (FES Section 5.14.1). Accordingly, the applicant will implement a Salt Drift Monitoring Program as discussed in Section 4.2.2 of this Plan.

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP*. Changes in station design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

Before engaging in additional construction or operational activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measurable nonradiological environmental effects are confined to the on-site areas previously disturbed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

* This provision does not relieve the licensee of the requirements of 10 CFR 50.59.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level; or (3) a matter, not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of the Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

3.2 Reporting Related to the NJPDES Permit and State Certification

Changes to, or renewals of, the NJPDES Permits or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. 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.

The licensee shall notify the NRC of changes to the effective NJPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NJPDES Permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, and local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events; onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; fish kills; increase in nuisance organisms or conditions; and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decisions made by the State of New Jersey under the authority of the Clean Water Act and, in the case of sea turtles and shortnose sturgeon, decisions made by the National Marine Fisheries Service (NMFS) under the authority of the Endangered Species Act for any requirements pertaining to aquatic monitoring.

4.2.2 Terrestrial Ecology Monitoring

PSE&G has completed the implementation of the Salt Drift Monitoring Program to assess the impacts of cooling tower salt drift on the environment in the HCGS vicinity. This study was completed by the submission of two reports: "Pre-operational Summary Report for Hope Creek Generating Station Salt Drift Monitoring Program, August 1984-December 1986" and "Operational Summary Report for Hope Creek Generating Station Salt Drift Monitoring Program, January 1987-March 1989". The pre-operational report was submitted to the NRC on April 30, 1987 (NLR-E87144) as an Appendix to the 1986 Annual Environmental Operating Report. The operational report was submitted to the NRC on October 10, 1989 (NLR-N89201).

The "Operational Summary Report" contained information that fulfilled the requirements of a final report, and therefore will be considered the "Final Report". This report discusses salt deposition data, native vegetation studies, comparison of estimated salt drift and deposition with actual data, environmental effects of salt drift and pre- and post-operational data comparison.

The study indicated that only minor, localized effects of cooling tower drift deposition are occurring. Higher deposition rates potentially attributable to the cooling tower were measured at only one location, which is on station property at a distance of 0.4 km southeast of the cooling tower. The salt deposition rate at this site is 113 mg/m^2/month, which is well below the deposition levels that have been reported to cause vegetative damage of 10,000 mg/m^2/year. Hope Creek Generating Station is surrounded by extensive areas of tidal salt marsh and the nearest uplands are located approximately three miles to the east, therefore no significant adverse impacts will occur as a result of cooling tower operation.

PSE&G has satisfied the commitments under this requirement. No further monitoring is required.

5.0 Administrative Procedures

5.1 Review

The licensee shall provide for review of compliance with the EPP. The review shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review function and results of the review activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the station. All other 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.

5.3 Changes in Environmental Protection Plan

Requests for changes 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.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of this EPP for the report period, including a comparison with related preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends toward irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of mitigating action.

The Annual Environmental Operating Report shall also include:

(1) A list of EPP noncompliances and the corrective actions taken to remedy them.

(2) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental question.

(3) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing results shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall: (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact, and plant operating characteristics; (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 with a copy of such report at the same time it is submitted to the other agency.

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. NPF-57

PSEG Nuclear LLC shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
97	The Licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated January 11, 1996, as supplemented by letters dated February 26, May 22, June 27, July 12, December 23, 1996, and March 17, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from March 21, 1997
103	The licensee shall relocate the list of "Motor Operated Valves - Thermal Overload Protection (BYPASSED)" from the Technical Specifications (Table 3.8.4.2-1) to the Updated Final Safety Analysis Report, as described in the licensee's application dated July 7, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from September 16, 1977.
105	The licensee shall use the Banked Pattern Withdrawal System or an improved version such as the Reduced Notch Worth Procedure as described in the licensee's application dated June 19, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from September 30, 1997.
110	The licensee shall relocate the suppression chamber water volume, as contained in Technical Specifications 3.5.3.a, 3.5.3.b, 3.6.2.1.a.1 and 5.2.1 to the Updated Final Safety Analysis Report, as described in the licensee's application dated August 20, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from November 6, 1977
114	The licensee is authorized to perform single cell charging of connected cells in OPERABLE class 1E batteries as described in the licensee's application dated September 8, 1998, as supplemented by letter dated December 8, 1998, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from February 9, 1999.

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