UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PSEG NUCLEAR LLC

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

FACILITY OPERATING LICENSE

Renewed License No. DPR-70

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

   A. The application for a renewed license, filed by PSEG Nuclear LLC acting on its own behalf and as agent for Exelon Generation Company, LLC* complies with the standards and requirements of the Atomic Energy Act (the Act) of 1954, as amended, and the Commission's rules and 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 Salem Nuclear Generating Station, Unit No. 1 (facility) has been substantially completed in conformity with Provisional Construction Permit No. CPPR-52 and the application, as amended, the provisions of the Act and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   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 rules and regulations of the Commission;

   E. PSEG Nuclear LLC is technically qualified and the licensees are financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

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

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* The Commission approved a transaction on November 16, 2021, that resulted in Exelon Generation Company, LLC being renamed Constellation Energy Generation, LLC. References to "the licensees" are to PSEG Nuclear LLC and Constellation Energy Generation, LLC.

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. DPR-70, subject to the conditions for protection of the environment set forth in the Technical Specifications, 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 including 10 CFR Sections 30.33, 40.32, and 70.23 and 70.31; and

J. 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 have been identified to 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, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

2. Renewed Facility Operating License No. DPR-70 is hereby issued to PSEG Nuclear LLC and Constellation Energy Generation, LLC (the licensees) to read as follows:

A. This renewed license applies to the Salem Nuclear Generating Station, Unit No. 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by PSEG Nuclear LLC and Constellation Energy Generation, LLC and operated by PSEG Nuclear LLC. The facility is located on the applicants' site in Salem County, New Jersey, on the southern end of Artificial Island on the east bank of the Delaware River in Lower Alloways Creek Township, and is described in the "Final Safety Analysis Report" as supplemented and amended and the Environmental Report as supplemented and amended.

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

(1) PSEG Nuclear LLC and Constellation Energy Generation, LLC to possess the facility at the designated location in Salem County, New Jersey, in accordance with the procedures and limitations set forth in this renewed license;

(2) PSEG Nuclear LLC, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use and operate the facility;

(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 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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) <u>Maximum Power Level</u>

PSEG Nuclear LLC is authorized to operate the facility at a steady state reactor core power level not in excess of 3459 megawatts (one hundred percent of rated core power).

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 347, 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) Deleted Per Amendment 22, 11-20-79

(4) <u>Less than Four Loop Operation</u>

PSEG Nuclear LLC shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this renewed license) with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensees and approval for less than four loop operation at power levels above P-7 has been granted by the Commission by Amendment of this renewed license.

(5) PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety

Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, 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.

(6) The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for recording and management of data;

5. Procedures defining corrective actions for off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

(7) <u>Systems Integrity</u>

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(8) Iodine Monitoring

The licensee shall implement 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.

(9) Backup Method for Determining Subcooling Margin

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.

(10) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 341, are hereby incorporated into this renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Additional Conditions.

(11) DELETED

(12) Constellation Energy Generation, LLC shall provide to the Director of the Office of Nuclear Reactor Regulation or the Director of the Office of Nuclear Material Safety and Safeguards, as applicable, a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from Constellation Energy Generation, LLC to its direct or indirect parent, or to any other affiliated company, facilities for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of Constellation Energy Generation, LLC's consolidated net utility plant, as recorded on Constellation Energy Generation, LLC's books of account.

(13) DELETED

(14) DELETED

(15) DELETED

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(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) Upon implementation of Amendment No. 286 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 4.7.6.2, in accordance with TS 6.18.c.(i), the assessment of CRE habitability as required by Specification 6.18.c.(ii). and the measurement of CRE pressure as required by Specification 6.18.d, shall be considered met. Following implementation:

a. The first performance of SR 4.7.6.2. in accordance with Specification 6.18.c.(i), shall be within the specified frequency of 6 years, plus the 18 month allowance of SR 4.0.2, as measured from June 4, 2003, 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 6.18.c.(ii), shall be 3 years, plus the 9 month allowance of SR 4.0.2, as measured from June 4, 2003, 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 6.18.d, shall be within 18 months, plus the 138 days allowed by SR 4.0.2, as measured from September 22, 2005, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.

(18) 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 PSEG Nuclear LLC evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(19) Appendix A of NUREG-2101, "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station," dated June 2011, and PSEG Nuclear LLC UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on May 18, 2011, describe certain future programs and activities to be completed before the period of extended operation. PSEG Nuclear LLC shall complete these activities no later than August 13, 2016, and shall notify the NRC in writing when implementation of these activities is complete.

(20) All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the

specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC. Changes to the withdrawal schedule or storage requirements shall be submitted to the NRC as a report in accordance with 10 CFR 50.4.

(21) PSEG Nuclear LLC shall take one core sample in the Unit 1 spent fuel pool west wall, by the end of 2013, and one core sample in the east wall where there have been indications of borated water ingress through the concrete, by the end of 2015. The core samples (east and west walls) will expose the rebar, which will be examined for signs of corrosion. Any sample showing signs of concrete degradation and/or rebar corrosion will be entered into the licensee's corrective action program for further evaluation. PSEG Nuclear LLC shall submit a report in accordance with 10 CFR 50.4 no later than three months after each sample is taken on the results, recommendations, and any additional planned actions.

(22) Concurrent with the first use of the chilled water cross-tie as allowed by Technical Specification 3.7.10c, PSEG shall confirm the required performance of the chilled water system cross-tie.

D. Paragraph 2.D. has been combined with paragraph 2.E. per Amendment No. 86, June 27, 1988.

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 Plan," "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. 300 as supplemented by changes approved by License Amendment Nos. 302, 306, and 318.

F. In accordance with the requirement imposed by the October 8, 1976, order of the United States Court of Appeals for the District of Columbia Circuit in Natural Resources Defense Council v. Nuclear Regulatory Commission, No. 74-1385 and 74-1586, that the Nuclear Regulatory Commission "shall make any licenses granted between July 21, 1976 and such time when the mandate is issued subject to the outcome of the proceedings herein," the license amendment issued herein shall be subject to the outcome of such proceedings.

G. Prior to startup following the first regularly scheduled refueling outage, Public Service Electric and Gas Company shall install, to the satisfaction of the Commission, a long-term means of protection against reactor coolant system over-pressurization when water-solid.

H. This renewed license is effective as of the date of its issuance. Renewed Facility Operating License No. DPR-70, as amended, shall expire at midnight, August 13, 2036.

I. IAEA SAFEGUARDS

1. INCORPORATION OF FACILITY ATTACHMENT:

Pursuant to 10 CFR 75.8, NRC License No. DPR-70 is hereby amended to incorporate by reference Codes 1. through 7. of Facility Attachment No.13 dated October 1, 1986, to the US/IAEA of Safeguards Agreement.

2. FACILITY ATTACHMENT CODE 2.2

Notification of the changes referred to in Code 2.2 of the facility attachment is the responsibility of the operating facility. They can be notified to the NRC with a Concise Note (DOE/NRC Form 740M) or a letter. Notification is required 70 days prior to the event.

3. FACILITY ATTACHMENT CODE 3.1.3 & 5.1.2 & 5.2.3

The itemized lists of nuclear material to be provided to the IAEA as of cycle shutdown date prior to physical inventory taking are:

1. A complete list of fuel assemblies by ID number at all locations.

2. Reactor and fuel storage maps showing location of fuel by ID number at time of physical inventory taking.

3. A list, by batch, of any other accountable nuclear material, e.g., start-up sources, samples.

4. FACILITY ATTACHMENT CODE 3.2.2

Please refer to NRC letter dated May 27, 1986, to Mr. C.A. McNeill from Steven A. Varga which spells out timeliness and procedures for notification under this code.

5. FACILITY ATTACHMENT CODE 5.1.1 & 6.1.1

The statement "when calculated" means at least as often as required on page 2 of NUREG/BR-0006 Revision 2 or more often, at your option, if you calculate burn up more than every six months.

6. FACILITY ATTACHMENT CODE 6.1.1 & 6.1.2

The phrase "as specified in relevant paragraphs of Code 10" is a requirement on the U.S. All of the paragraphs in the US/IAEA Agreement that require a report from the U.S. to the IAEA based on source data from an operating facility have been incorporated into NUREG's BR-0006 and 0007 so that the NRC may collect the needed data for transmittal to the IAEA. PSEG Nuclear LLC should follow these NUREGs precisely in reporting inventory changes. A complete response to the reporting instructions in the NUREGs will satisfy the requirements specified in Code 10.

7. FACILITY ATTACHMENT CODE 6.2.2

The phrase "precise forecasts" means best estimates. These required concise notes should be dispatched to the NRC at least 40 days in advance of a projected 6 month operational programming.

8. FACILITY ATTACHMENT CODE 6.3.1 & 6.3.2

See response to Code 6.1.1 and 6.1.2 above.

9. FACILITY ATTACHMENT CODE 7.9

The specific facility health and safety rules and regulations to be observed by the Agency's (IAEA) inspectors, as specified in Paragraph 54 of the design information as of October 10, 1986, provided by the U.S.A. mean:

Agency inspectors who have previously visited the facility will be informed as necessary at the time of entry into the facility of health and safety rules and ad hoc rules as might be required in view of a special situation that has occurred at the facility since the inspector's last visit to the facility. The briefing will be of a short duration, not to exceed 30 minutes, covering topics deemed relevant by the licensee.

Agency inspectors who have not previously visited the facility will be informed as necessary at the time of entry into the facility of health and safety rules and ad hoc rules as might be required in view of a special situation that has occurred at the facility. The briefing will be of an appropriate duration, not to exceed three hours, and consist of topics deemed relevant by the licensee.

In either case, the licensee should take into account the Agency inspector's prior training, expertise and experience. In neither case shall the Agency inspector be subject to any form of evaluation or testing by facility representatives or representatives of the U.S. Government.

For health and safety reasons, Agency inspectors will be escorted by qualified facility personnel at times deemed appropriate by the licensee.

10. TERMINATION

Pursuant to the provisions of 10 CFR 75.41, the Commission will inform the licensee, in writing, when its installation is no longer subject to Article 39(b) of the principal text of the US/IAEA Safeguards Agreement. The IAEA Safeguards License Conditions incorporating Code 7. of the Facility Attachment as part of NRC License DPR-70 will be terminated as of the date of such notice from the Commission. However, since the IAEA may elect to maintain the licensee's installation under Article 2(a) of the Protocol, provisions equivalent to Codes 1. through 6. of the Facility Attachment (with possible appropriate modifications) may still apply, and accordingly all other IAEA Safeguards License Conditions to NRC License No. DPR-70 will remain in effect until the Commission notifies the licensee otherwise. If this option is not selected by the IAEA, the Commission will then notify the licensee that all License Conditions pertaining to the US/IAEA Safeguards Agreement are terminated.

J. RELOCATED TECHNICAL SPECIFICATIONS

PSEG Nuclear LLC shall relocate certain technical specification requirements to licensee-controlled documents as described below. The location of these requirements shall be retained by the licensee.

a. This license condition approves the relocation of certain technical specification requirements to licensee-controlled documents (UFSAR), as described in the licensee's applications with the staff's safety evaluation approval and Amendment No. as noted below:

Licensee's Applications	Safety Evaluations	Amendment Nos.
September 25, 1996	January 30, 1997	189

Implementation shall include the relocation of technical specifications requirements to the appropriate licensee-controlled document as identified in the licensee's application.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A – Technical Specifications
2. Appendix B – Environmental Protection Plan
3. Appendix C – Additional Conditions

Date of Issuance: June 30, 2011

SALEM NUCLEAR GENERATING STATION
UNIT 1
TECHNICAL SPECIFICATIONS

APPENDIX "A" TO LICENSE NO. DPR-70

AUGUST 13, 1976

ISSUED BY THE UNITED STATES NUCLEAR REGULATORY COMMISSION

Trans w/original
License, dtd
8-13-76

SALEM NUCLEAR GENERATING STATION

UNIT 1

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

LICENSE NO. DPR-70

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

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

AXIAL FLUX DIFFERENCE

1.3 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass 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. Whenever an RTD or thermocouple sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

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 the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or 0-rings) is OPERABLE.

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT (COLR) 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. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of 1-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using the “Thyroid” Committed Dose Equivalent (CDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion.”

DOSE EQUIVALENT XE-133

1.11 DOSE EQUIVALENT XE-133 shall be that concentration of XE-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides KR-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135 and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at a minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, “External Exposure to Radionuclides in Air, Water and Soil.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 230 steps withdrawn, inclusive. FULLY WITHDRAWN will be specified in the current reload analysis.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except Reactor Coolant Pump Seal Water Injection) into closed systems, such as pump seal or valve packing leaks that are 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 leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system (primary-to-secondary leakage).

INSERVICE TESTING PROGRAM

1.15.1 The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

1.16 Not Used

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION 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 and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8 respectively.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.20 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 Updated FSAR, 2) authorized under the provisions of 10CFR50.59, or 3) otherwise by the Commission.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.20a The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the Overpressure Protection System setpoint and enable temperature, for the current reactor vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Technical Specification Section 6.9.1.11.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging 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 radioactive waste.

PURGE - PURGING

1.23 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 a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be FULLY WITHDRAWN.

1.29 Not Used

SOLIDIFICATION

1.30 Not Used

SOURCE CHECK

1.31 SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to either (a) an external source of increased radioactivity, or (b) an internal source of radioactivity (keep-alive source), or (c) an equivalent electronic source check.

STAGGERED TEST BASIS

1.32 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

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

UNIDENTIFIED LEAKAGE

1.34 UNIDENTIFIED LEAKAGE shall be all leakage (except Reactor Coolant Pump Seal Water Injection) which is not IDENTIFIED LEAKAGE.

1.35 Not Used

VENTILATION EXHAUST TREATMENT SYSTEM

1.36 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine and 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

1.37 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

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

* Excluding decay heat.

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

TABLE 1.2

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 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Prior to each release.
N.A.	Not applicable.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

-to be implemented prior to entry into Mode 2 from the current outage

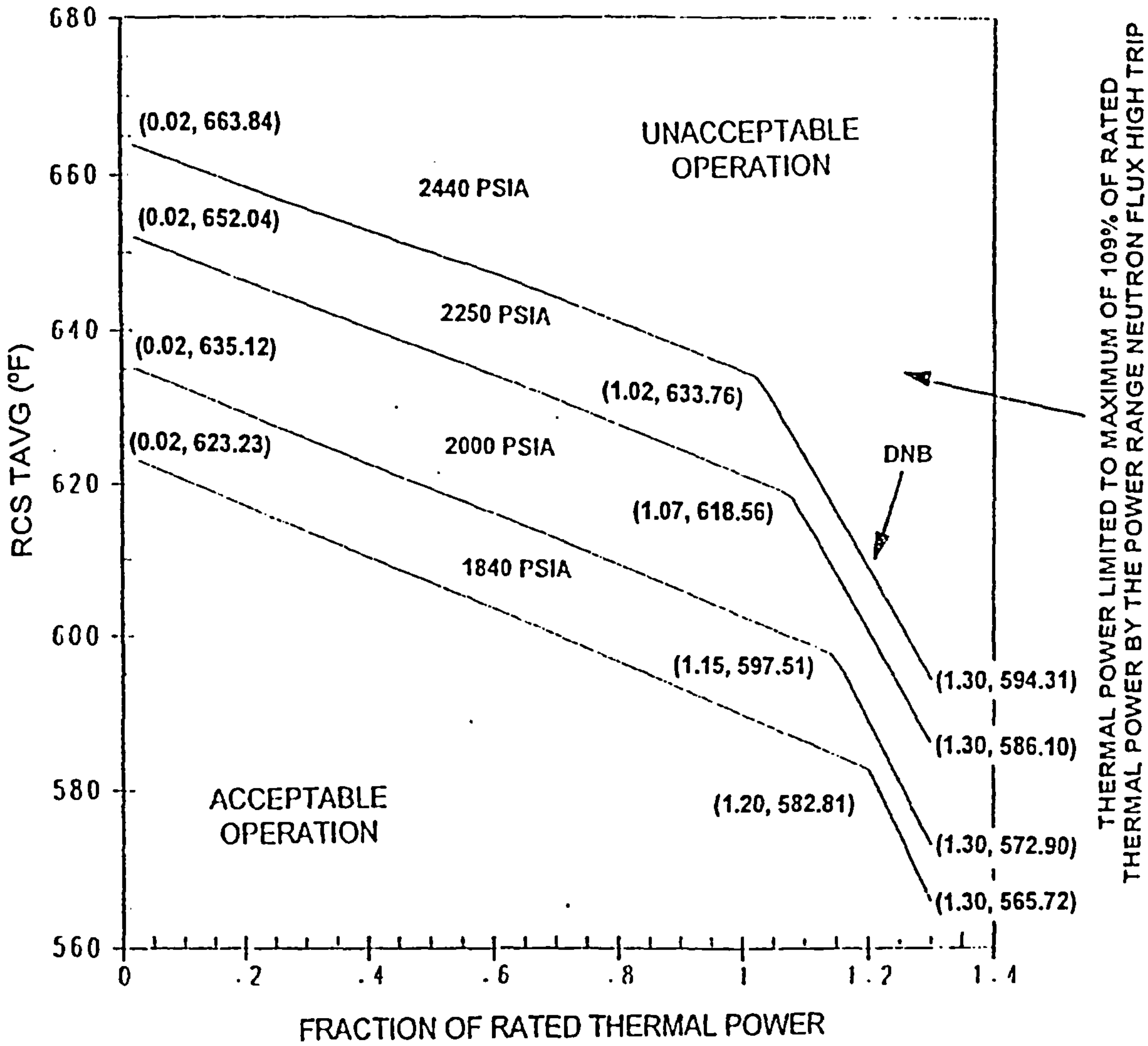

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

-to be implemented prior to entry into Mode 2 from the current outage

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Deleted		
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 38.5% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.44 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

* Design flow is 82,500 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 14.0% of narrow range instrument span-each steam generator	≥ 13.0% of narrow range instrument span-each steam generator
14. Deleted		
15. Undervoltage-Reactor Coolant Pumps	≥ 2900 volts-each bus	≥ 2850 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 56.5 Hz - each bus	≥ 56.4 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 45 psig
B. Turbine Stop Valve Closure	≤ 15% off full open	≤ 15% off full open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 30 secs ±10%, τ_2 = 4 secs. ±10%

S = Laplace transform operator, Sec^{-1}

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops

K1 = 1.22
K2 = 0.02037
K3 = 0.001020

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between -33 percent and +11 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of (q_t - q_b) exceeds -33 percent, the ΔT trip setpoint shall be automatically reduced by 2.34 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of (q_t - q_b) exceeds +11 percent, the ΔT trip setpoint shall be automatically reduced by 2.37 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: Overpower: $\Delta T \leq \Delta T_o \; [K_4 - K_5\left[\frac{\tau_3 S}{1 + \tau_3 S}\right]T - K_6(T - T") - f_2(\Delta I)]$

where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T" = Indicated T_{avg} at RATED THERMAL POWER ≤ 577.9°F

K_4 = 1.09

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.00149/°F for T > T"; K_6 = 0 for T ≤ T"

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs. ±10%

S = Laplace transform operator, Sec^{-1}

$f_2(\Delta I)$ = 0 for all ΔI

NOTE 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.1 percent.

NOTE 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.1 percent.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 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 MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

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 a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY 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.

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

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

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE 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 MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0.5 DELETED

3.0.6 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.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES 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 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 the 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. The delay period is only applicable when there is a reasonable expectation the Surveillance will be met when performed. 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 a MODE 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 a MODE 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 MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} > 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ 1.3% Δk/k.

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

ACTION:

With the SHUTDOWN MARGIN < 1.3% Δk/k, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ 1.3% Δk/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, in accordance with the Surveillance Frequency Control Program by verifying that control banks are within the limits in the COLR per Specification 3.1.3.5.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits in the COLR per specification 3.1.3.5.

---

* See Special Test Exception 3.10.1

\# With Keff ≥ 1.0

\## With Keff < 1.0

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit in the COLR per Specification 3.1.3.5.

e. When in MODES 3 or 4, in accordance with the Surveillance Frequency Control Program by consideration of the following factors:

   1. Reactor coolant system boron concentration,

   2. Control rod position,

   3. Reactor coolant system average temperature,

   4. Fuel burnup based on gross thermal energy generation,

   5. Xenon concentration, and

   6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% $\Delta k/k$ in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be ≥ 1.0% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN < 1.0% Δk/k, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be ≥ 1.0% Δk/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. In accordance with the Surveillance Frequency Control Program by consideration of the following factors:

    1. Reactor coolant system boron concentration,

    2. Control rod position,

    3. Reactor coolant system average temperature,

    4. Fuel burnup based on gross thermal energy generation,

    5. Xenon concentration, and

    6. Samarium concentration.

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less positive than or equal to 0 Δk/k/°F.

APPLICABILITY: Beginning of Cycle Life (BOL) Limit - MODES 1 and 2* only#
End of Cycle Life (EOL) Limit - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operations in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits in the COLR per Specification 3.1.3.5.

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

   3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

* With K_{eff} greater than or equal to 1.0

\# See Special Test Exception 3.10.3

-to be implemented prior to entry into Mode 2 from the current outage

NOV 2 6 1997

MODERATOR TEMPERATURE COEFFICIENT

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

– to be implemented prior to entry into Mode 2 from the current outage NOV 2 6 1997

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 541°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 541°F, restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 551°F with the T_{avg} - T_{ref} Deviation Alarm not reset.

#With $K_{eff} \geq 1.0$.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage system is OPERABLE, per Specification 3.1.2.6a while in MODE 4, or per Specification 3.1.2.5a while in MODE 5 or 6, or

b. A flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank is OPERABLE per Specification 3.5.5 while in MODE 4, or per Specification 3.1.2.5b while in MODE 5 or 6.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. When the boric acid tank is a required water source, by verifying in accordance with the Surveillance Frequency Control Program that:

(1) The flow path from the boric acid tank to the boric acid transfer pump, the boric acid transfer pump, and the recirculation path from the boric acid transfer pump to the boric acid tank is ≥ 63°F, and

(2) The flow path between the boric acid transfer pump recirculation line to the charging pump suction line is ≥ 50°F,

b. 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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. By verifying in accordance with the Surveillance Frequency Control Program that:

(1) The flow path from the boric acid tank to the boric acid transfer pump and from the recirculation line back to the boric acid tank is ≥ 63°F, and

(2) the flow path between the boric acid tank recirculation line to the charging pump suction line is ≥ 50°F,

b. 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.

SURVEILLANCE REQUIREMENTS (Continued)

c. In accordance with the Surveillance Frequency Control Program during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

d. In accordance with the Surveillance Frequency Control Program by verifying that the flow path required by specification 3.1.2.2.a delivers at least 33 gpm to the Reactor Coolant System.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.#

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

\# A maximum of one centrifugal charging pump shall be OPERABLE while in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, MODE 5, or MODE 6 when the head is on the reactor vessel.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

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BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum contained volume of 2,600 gallons,
2. Between 6,560 and 6,990 ppm of boron, and,
3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 37,000 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. For the boric acid storage system, when it is the source of borated water in accordance with the Surveillance Frequency Control Program by:

1. Verifying the boron concentration of the water,
2. Verifying the water level of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

SURVEILLANCE REQUIREMENTS (Continued)

b. For the refueling water storage tank by:

1. Verifying the boron concentration in accordance with the Surveillance Frequency Control Program,

2. Verifying the borated water volume in accordance with the Surveillance Frequency Control Program, and

3. Verifying the solution temperature in accordance with the Surveillance Frequency Control Program, when it is the source of borated water and the outside air temperature is less than 35°F.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by specifications 3.1.2.1 and 3.1.2.2:

a. A boric acid storage system with:

1. A contained volume of borated water in accordance with figure 3.1.2,

2. A boron concentration in accordance with figure 3.1-2, and

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank per Specification 3.5.5.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required boration water systems, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta K/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, perform the Action described in Specification 3.5.5.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

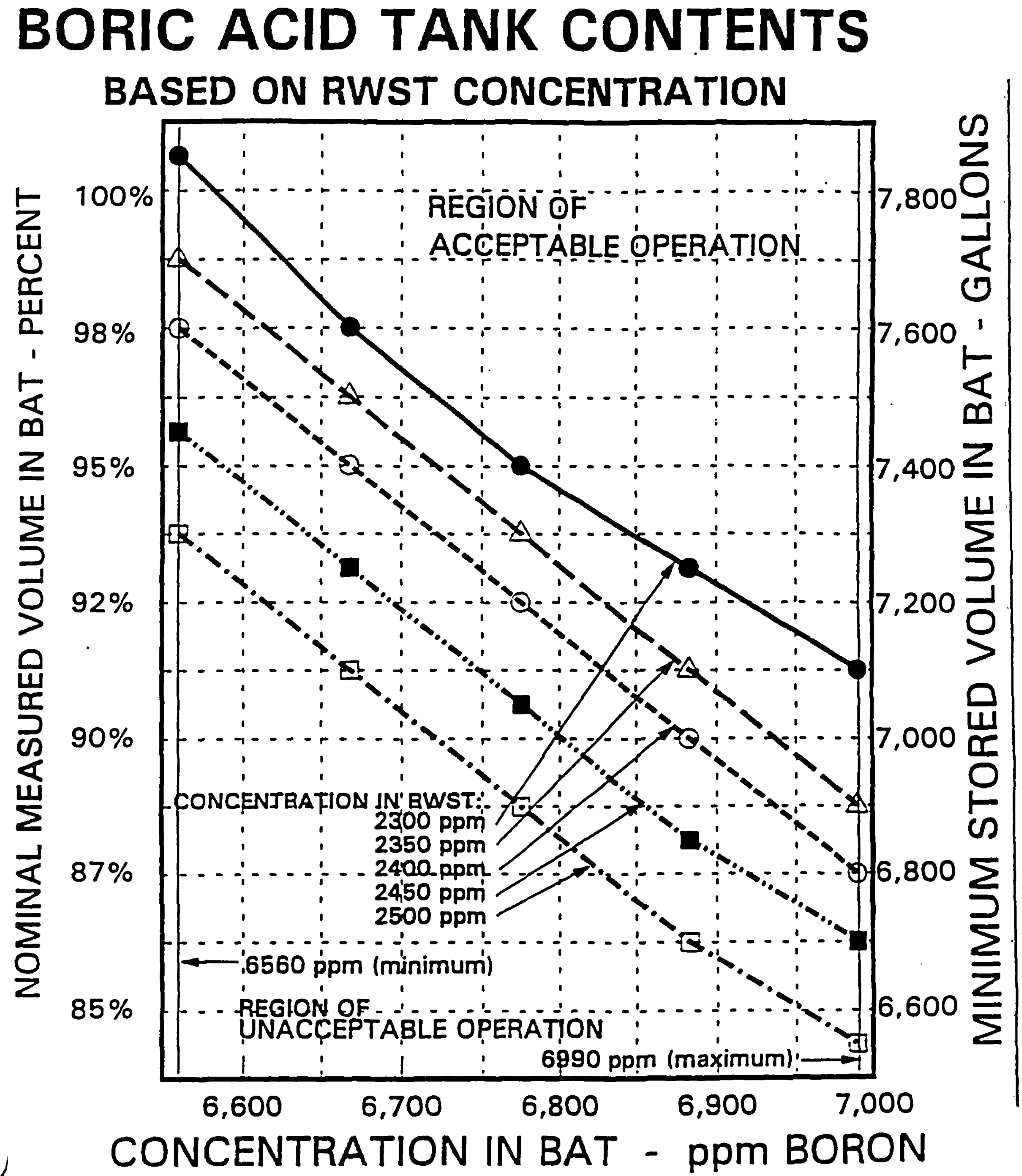
a. For the boric acid storage system, when it is the source of borated water in accordance with the Surveillance Frequency Control Program by:

   1. Verifying the boron concentration in each water source,

   2. Verifying the water level of each water source, and

   3. Verifying the boric acid storage system solution temperature.

b. For the refueling water storage tank per Surveillance 4.5.5.



BORIC ACID TANK CONTENTS
BASED ON RWST CONCENTRATION
REGION OF ACCEPTABLE OPERATION
NOMINAL MEASURED VOLUME IN BAT - PERCENT
100%
98%
95%
92%
90%
87%
85%
MINIMUM STORED VOLUME IN BAT - GALLONS
7,800
7,600
7,400
7,200
7,000
6,800
6,600
CONCENTRATION IN RWST:
2300 ppm
2350 ppm
2400 ppm
2450 ppm
2500 ppm
6560 ppm (minimum)
REGION OF UNACCEPTABLE OPERATION
6990 ppm (maximum)
6,600
6,700
6,800
6,900
7,000
CONCENTRATION IN BAT - ppm BORON

Figure 3.1-2

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, shall be OPERABLE and positioned within ± 18 steps (indicated position) when reactor power is ≤ 85%, RATED THERMAL POWER, or ± 12 steps (indicated position) when reactor power is > 85% RATED THERMAL POWER, of their group step counter demand position within one hour after rod motion.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or mis-aligned from the group step counter demand position by more than ± 18 steps(indicated position) at ≤ 85 % RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER, be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or mis-aligned from its group step counter demand position by more than ± 18 steps (indicated position) at ≤ 85 % RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER, POWER OPERATION may continue provided that within one hour either:

    1. Deleted

    2. The remainder of the rods in the bank with the inoperable rod are aligned to within ± 18 steps(indicated position) at ≤ 85 % RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER of the inoperable rod while maintaining the rod sequence and insertion limits in the COLR per specification 3.1.3.5. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation, or

    3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

---

* See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

a) A reevaluation of each accident analysis of table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

c) A core power distribution measurement is obtained and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.c.3.a and 3.1.3.1.c.3.c above are demonstrated.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod* shall be determined to be within the limits established in the limiting condition for operation in accordance with the Surveillance Frequency Control Program (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction in accordance with the Surveillance Frequency Control Program.

* Not required to be performed for rods associated with inoperable rod position indicator or demand position indicator.

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TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Mis-alignment

Loss Of Reactor Coolant From Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2.1 The shutdown and control rod position indication (RPI) systems shall be OPERABLE and capable of determining the actual and demanded rod positions as follows:

a. Analog rod position indicators;

   All Shutdown Banks: ± 18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-230 steps.

   Control Bank A: ± 18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-230 steps.

   Control Bank B: ± 18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 160-230 steps.

   Control Bank C and D: ± 18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-230 steps.

------------------------------Note------------------------------

Individual RPIs are not required to be OPERABLE for 1 hour following movement of the associated rods.

----

b. Group demand counters; ± 2 steps of the pulsed output of the Slave Cycler Circuit over the withdrawal range of 0-230 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one analog rod position indicator per group inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% RTP or the power distribution monitoring system is inoperable) at least once per 8 hours, or

   2. Verify the position of the non-indicating rod(s) indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% or the power distribution monitoring system is inoperable) within 8 hours and once per 31 EFPD thereafter, and 8 hours after discovery of each unintended rod movement, and 8 hours after each movement of the non-indicating rod(s) greater than 12 steps, and prior to THERMAL POWER exceeding 50% RTP, and 8 hours after reaching RTP, or

LIMITING CONDITION FOR OPERATION (Continued)

3. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With two or more analog rod position indicators per group inoperable:

1. Immediately place the control rods in manual control, and

2. Deleted

3. Verify the position of the rods with inoperable position indicators indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% RTP or the power distribution monitoring system is inoperable) at least once per 8 hours, and

4. Within 24 hours restore the inoperable rod position indicators to OPERABLE status such that a maximum of one rod position indicator per group is inoperable, or

5. Be in HOT STANDBY within the next 6 hours.

c. When one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position:

1. Determine the position of the non-indicating rod(s) indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% RTP or the power distribution monitoring system is inoperable) within 8 hours, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

d. With a maximum of one group demand position indicator per bank inoperable either:

1. Verify that all analog rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 18 steps when reactor power is ≤ 85% RATED THERMAL POWER or if reactor power is > 85% RATED THERMAL POWER, 12 steps of each other at least once per 8 hours, or

2. Reduce THERMAL POWER to less than 50% of RATED POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2.1.1 Each analog rod position indicator* shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 18 steps when reactor power is ≤ 85% RATED THERMAL POWER or if reactor power is > 85% RATED THERMAL POWER, 12 steps (allowing for one hour thermal soak after rod motion) in accordance with the Surveillance Frequency Control Program except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.

4.1.3.2.1.2 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

**Not required to be met for RPIs associated with rods that do not meet LCO 3.1.3.1*

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 230 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 & 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. In accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

---Note---
Not applicable to shutdown banks inserted while performing SR 4.1.3.1.2

APPLICABILITY: MODES 1*, and 2*#@

ACTION:

1. With one shutdown bank inserted ≤ 10 steps beyond FULLY WITHDRAWN; within 1 hour verify all control banks are within the insertion limits specified in the COLR and determine the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied; and within 24 hours restore the shutdown bank to FULLY WITHDRAWN or be in HOT STANDBY within the next 6 hours.

2. With a maximum of one shutdown rod not FULLY WITHDRAWN, for reasons other than Action 1, within one hour either:

    a. FULLY WITHDRAW the rod, or,

    b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators**:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. In accordance with the Surveillance Frequency Control Program thereafter.

---

* See Special Test Exceptions 3.10.2 and 3.10.3

** Not required to be performed until 1 hour after associated rod motion.

With Keff greater than or equal to 1.0

@ Surveillance 4.1.3.4.a is applicable prior to withdrawing control banks in preparation for startup (Mode 2).

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

--Note--
Not applicable to control banks inserted while performing SR 4.1.3.1.2

APPLICABILITY: MODES 1*, and 2*#

ACTION:

1. With control bank A, B, or C inserted ≤ 10 steps beyond the insertion limits specified in the COLR; within 1 hour verify all shutdown banks are FULLY WITHDRAWN and determine the SHUTDOWN MARGIN requirement of 3.1.1.1 is satisfied; and within 24 hours restore the control bank to within the insertion limits specified in the COLR or be in HOT STANDBY within the next 6 hours.

2. With the control banks inserted beyond the above insertion limits, for reasons other than Action 1, either:

   a. Restore the control banks to within the limits within two hours, or

   b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

   c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

--Note--
Not required to be performed until 1 hour after associated rod motion.

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits in accordance with the Surveillance Frequency Control Program by use of the group demand counters and verified by the analog rod position indicators except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

---

* See Special Test Exceptions 3.10.2 and 3.10.3

\# With K_{eff} greater than or equal to 1.0

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AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band about the target flux difference as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR and with THERMAL POWER:

   1. Above 90% of RATED THERMAL POWER, within 15 minutes:

      a) Either restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

   2. Between 50% and 90% of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the limits specified in the COLR for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

      b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

---

*See Special Test Exception 3.10.2

- to be implemented prior to entry into Mode 2 from the current outage

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR and ACTION a.2.a)1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the limits specified in the COLR for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. In accordance with the Surveillance Frequency Control Program when the AFD Monitor Alarm is OPERABLE, and

2. In accordance with the Surveillance Frequency Control Program for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel in accordance with the Surveillance Frequency Control Program for the first 24 hours and in accordance with the Surveillance Frequency Control Program thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside of the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the limits at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the limits at THERMAL POWER levels below 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference shall be determined by measurement in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.

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-to be implemented prior to entry into mode 2
from the current outage

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(z)$ shall be limited by the following relationships:

$$F_Q(z) \le \frac{F^{RTP}_Q}{P} * K(z) \text{ for } P > 0.5, \text{ and}$$

$$F_Q(z) \le \frac{F^{RTP}_Q}{0.5} * K(z) \text{ for } P \le 0.5,$$

Where: F^{RTP}_Q = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}, \text{ and}$$

K(z) = the normalized $F_Q(z)$ as a function of core height as specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through a core power distribution measurement to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map:

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the Measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS or the moveable incore detectors when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1 + PF_{xy} (1-P)]$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} in the COLR, and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

SURVEILLANCE REQUIREMENTS (Continued)

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional core power distribution measurements shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L}:

    a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

    b) In accordance with the Surveillance Frequency Control Program, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional core power distribution measurements shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} in accordance with the Surveillance Frequency Control Program.

e. The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the COLR per specification 6.9.1.9.

f. The F_{xy} limits of e, above, are not applicable to the excluded core plane regions as measured in percent of core height from the bottom of the fuel.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a core power distribution measurement and increased by the applicable manufacturing and measurement uncertainties as specified in the COLR.

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2 from the current outage

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H} [1.0 + PF_{\Delta H} (1.0 - P)]$$

Where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the COLR, and

P is $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru a core power distribution measurement that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through a core power distribution measurement to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by obtaining a core power distribution measurement:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. In accordance with the Surveillance Frequency Control Program.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by the applicable $F^N_{\Delta H}$ uncertainties specified in the COLR.

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but $\leq$ 1.09:

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio in accordance with the Surveillance Frequency Control Program when the alarm is OPERABLE.

b. Calculating the ratio in accordance with the Surveillance Frequency Control Program during steady state operation when the alarm is inoperable.

c. Obtaining a core power distribution measurement to determine the QUADRANT POWER TILT RATIO in accordance with the Surveillance Frequency Control Program when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits in accordance with the Surveillance Frequency Control Program.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within the limits of Table 3.2-1 by performing a precision heat balance within 24 hours after achieving steady state conditions ≥ 90% RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
	4 Loops In Operation
Reactor Coolant System T_{avg}	≤ 582.9°F
Pressurizer Pressure	≥ 2200 psia*
Reactor Coolant System Flow	≥ 341,000 gpm#

* Limit not applicable during either THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

\# Includes a 2.4% flow measurement uncertainty plus a 0.1% measurement uncertainty due to feedwater venturi fouling.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be verified to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2, and *	12
2. Power Range, Neutron Flux	4	2	3	1,2, and 3*	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1,2	2
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1[(a)],2[(b)]	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2[(c)]	4
B. Shutdown	2	0	1	3,4, and 5	5
C. Shutdown	2	1	2	3*,4*, and 5*	7
7. Overtemperature ΔT	4	2	3	1,2	6
8. Overpower ΔT	4	2	3	1,2	6
9. Pressurizer Pressure-Low	4	2	3	1,2	6
10. Pressurizer Pressure--High	4	2	3	1,2	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11.	Pressurizer Water Level--High	3	2	2	1, 2	6
12.	Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any oper- ating loop	2/loop in each oper- ating loop	1	6
13.	Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two oper- ating loops	2/loop in each oper- ating loop	1	6
14.	Steam Generator Water Level-- Low-Low	3/loop	2/loop in any oper- ating loops	2/loop in each oper- ating loop	1, 2	6
15.	Deleted					
16.	Undervoltage-Reactor Coolant Pumps	4-1/bus	1/2 twice	3	1	6
17.	Underfrequency-Reactor Coolant Pumps	4-1/bus	1/2 twice	3	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Turbine Trip					
a. Low Autostop Oil Pressure	3	2	2	1#	6
b. Turbine Stop Valve Closure	4	4	3	1#	6
19. Safety Injection Input from ESF	2	1	2	1,2	10
20. Reactor Coolant Pump Breaker Position Trip (above P-7)	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2	1	2	1,2 3*,4*,5*	1###,14 13
22. Automatic Trip Logic	2	1	2	1,2 3*,4*,5*	10 13

TABLE 3.3-1 (Continued)

TABLE NOTATION

(a) Below the P-10 (Power Range Neutron Flux) interlocks
(b) Above the P-6 (Intermediate Range Neutron Flux) interlocks
(c) Below the P-6 (Intermediate Range Neutron Flux) interlocks

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

** Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

Above the P-9 (Power Range Neutron Flux) interlock.

If ACTION Statement 1 is entered as a result of Reactor Trip Breaker (RTB) or Reactor Trip Bypass Breakers (RTBB) maintenance testing results exceeding the following acceptance criteria, NRC reporting shall be made within 30 days in accordance with Specification 6.9.2:

1. A RTB or RTBB trip failure during any surveillance test with less than or equal to 300 grams of weight added to the breaker trip bar.

2. A RTB or RTBB time response failure that results in the overall reactor trip system time response exceeding the Technical Specification limit.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel (RTB) to OPERABLE within 24 hours or be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours.

b. The Minimum Channels OPERABLE requirement is met; however, one channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to ≤ 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to ≤ 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

TABLE 3.3-1 (Continued)

ACTION 3 - With the number of channels OPERABLE:

a. One less than required by the Minimum Channels OPERABLE requirement

1. Reduce THERMAL POWER to < P-6 within 24 hours or,

2. Increase THERMAL POWER to > P-10 within 24 hours.

b. Two less than required by the Minimum Channels OPERABLE requirement

1. Immediately suspend operations involving positive reactivity additions** and,

2. Reduce THERMAL POWER to < P-6 within 2 hours.

ACTION 4 - With the number of channels OPERABLE:

a. One less than required by the Minimum Channels OPERABLE requirement, immediately suspend operations involving positive reactivity additions**.

b. Two less than required by the Minimum Channels OPERABLE requirement, immediately open reactor trip breakers.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1.1.

TABLE 3.3-1 (Continued)

ACTION 7 - With the number of channels OPERABLE:

a. One less than required by the Minimum Channels OPERABLE requirement:

1. Restore the channel to OPERABLE status within 48 hours or

2. Initiate action to fully insert all rods within 48 hours and place the Control Rod Drive System in a condition incapable of rod withdrawal within the next hour.

b. Two less than required by the Minimum Channels OPERABLE requirement, immediately open reactor trip breakers.

ACTION 8 - NOT USED

ACTION 9 - NOT USED

ACTION 10 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY in the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1, provided the other channel is OPERABLE.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 72 hours.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 14 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and be in at least HOT STANDBY within 6 hours. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels $< 4.7 \times 10^{-6}$ % of RTP.	P-6 prevents or defeats the manual block of source range reactor trip.
P-7	With 2 of 4 Power Range Neutron Flux Channels ≥ 11% of RATED THERMAL POWER or 1 of 2 Turbine steam line input pressure channels ≥ a pressure equivalent to 11% of RATED THERMAL POWER.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump undervoltage and under-frequency, pressurizer low pressure, pressurizer high level, and the opening of more than one reactor coolant pump breaker.
P-8	With 2 of 4 Power Range Neutron Flux channels ≥ 36% of RATED THERMAL POWER.	P-8 prevents or defeats the automatic block of reactor trip on low coolant flow in a single loop.
P-9	With 2 of 4 Power Range neutron flux channels ≥ 50% RATED THERMAL POWER.	P-9 prevents or defeats the automatic block of reactor trip on turbine trip.
P-10	With 3 of 4 Power range neutron flux channels < 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

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TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(15)	CHANNEL CALIBRATION(15)	CHANNEL FUNCTIONAL TEST(15)	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	(9)	1, 2, and *
2. Power Range, Neutron Flux		(2), (3) (6) (17)	(18)	1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	(6)	(18)	1, 2
4. Deleted				
5. Intermediate Range, Neutron Flux		(6), #, ##	S/U(1) #, ##	1(a), 2(b)
6. Source Range, Neutron Flux				
A. Startup		(6), #, ##	(16),(8) and S/U(1) #, ##	2(7)
B. Shutdown		(6)	N.A.	3, 4, 5
C. Shutdown		(6), #, ##	(10), #, ##	3*, 4*, 5*
7. Overtemperature ΔT				1, 2
8. Overpower ΔT				1, 2
9. Pressurizer Pressure--Low				1, 2
10. Pressurizer Pressure--High				1, 2
11. Pressurizer Water Level--High				1, 2
12. Loss of Flow - Single Loop				1

If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the nominal Trip Setpoint at the completion of the surveillance; otherwise the channel shall be declared inoperable. Setpoints more conservative than the nominal Trip Setpoint are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The methodologies used to determine the as-found and as-left tolerances are specified in the Technical Specification Bases.

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

(b) Above the P-6 (Intermediate Range Neutron Flux) interlocks

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(15)	CHANNEL CALIBRATION(15)	CHANNEL FUNCTIONAL TEST(15)	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow Two Loops			N.A.	1
14. Steam Generator Water Level--Low-Low				1, 2
15. DELETED				
16. Undervoltage - Reactor Coolant Pumps	N.A.			1
17. Underfrequency - Reactor Coolant Pumps	N.A.			1
18. Turbine Trip				
a. Low Autostop Oil Pressure	N.A.	N.A.	S/U(1)	1#
b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1#
19. Safety Injection Input from ESF	N.A.	N.A.	(4)(5)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.		1
21. Reactor Trip Breaker	N.A.	N.A.	(5)(11)(13) (14)	1, 2 and *
22. Automatic Trip Logic	N.A.	N.A.	(5)	1, 2 and *

Above the P-9 (Power Range Neutron Flux) Interlock

TABLE 4.3-1 (Continued)

NOTATION

* With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ 3 percent.

(4) - Manual SSPS functional input check in accordance with the Surveillance Frequency Control Program.

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

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - If not performed within the frequency of the Surveillance Frequency Control Program, perform the CHANNEL FUNCTIONAL TEST within 4 hours after reducing power below P-6.

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip mechanism for the Manual Reactor Trip Function.

The Test shall also verify OPERABILITY of the Bypass Breaker Trip circuits.

(10) - If not performed within the frequency of the Surveillance Frequency Control Program, perform the CHANNEL FUNCTIONAL TEST within 4 hours of entering MODE 3 from MODE 2.

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Reactor Trip Breaker Undervoltage and Shunt Trip mechanisms.

(12) - DELETED

(13) - Verify operation of Bypass Breakers Shunt Trip function from local pushbutton while breaker is in the test position prior to placing breaker in service.

(14) - Perform a functional test of the Bypass Breakers U.V. Attachment via the SSPS.

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

(16) - At the frequency specified in the Surveillance Frequency Control Program.

(17) - In MODES 1, and 2, the SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance.

(18) - The SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable to MSIV closure time testing. The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION					
a. Manual Initiation	2	1	2	1,2,3,4	18
b. Automatic Actuation Logic	2	1	2	1,2,3,4	13
c. Containment Pressure-High	3	2	2	1,2,3	19
d. Pressurizer Pressure-Low	3	2	2	1,2,3#	19
e. Differential Pressure Between Steam Lines - High	3/steam line	2/steam line any steam line	2/steam line	1,2,3##	19
f. Steam Flow in Two Steam Lines-High	2/steam line	1/steam line any 2 steam lines	1/steam line	1,2,3##	19
COINCIDENT WITH EITHER					
Tavg--Low-Low	1 Tavg/loop	1 Tavg in any 2 loops	1 Tavg in any 3 loops	1,2,3##	19
OR, COINCIDENT WITH					
Steam Line Pressure-Low	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 3 loops	1,2,3##	19

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. CONTAINMENT SPRAY					
a. Manual	2 sets of 2	1 set of 2	2 sets of 2	1,2,3,4	18
b. Automatic Actuation Logic	2	1	2	1,2,3,4	13
c. Containment Pressure--High-High	4	2	3	1,2,3	16
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1,2,3,4	18
2) From Safety Injection Automatic Actuation Logic	2	1	2	1,2,3,4	13

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Phase "B" Isolation					
1) Manual	2 sets of 2	1 set of 2	2 sets of 2	1,2,3,4	18
2) Automatic Actuation Logic	2	1	2	1,2,3,4	13
3) Containment Pressure--High-High	4	2	3	1,2,3	16
c. Containment Ventilation Isolation					
1) Manual	2	1	2	1,2,3,4	17
2) Automatic Actuation Logic	2	1	2	1,2,3,4	13
3) Containment Atmosphere Gaseous Radioactivity-High		per table 3.3-6			

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION					
a. Manual	2/steam line	1/steam line	1/operating steam line	1, 2[(a)], 3[(a)]	23
b. Automatic Actuation Logic	2***	1	2	1, 2[(a)], 3[(a)]	20
c. Containment Pressure--High-High	4	2	3	1, 2[(a)], 3[(a)]	16
d. Steam Flow in Two Steam Lines--High	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2[(a)], 3##[(a)]	19
COINCIDENT WITH EITHER					
Tavg--Low-Low	1 Tavg/loop	1 Tavg in any 2 loops	1 Tavg in any 3 loops	1, 2[(a)], 3##[(a)]	19
OR, COINCIDENT WITH					
Steam Line Pressure-Low	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 3 loops	1, 2[(a)], 3##[(a)]	19

[(a)] Except when all MSIVs are closed.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Safety Injection	Refer to Functional Unit 1 for all initiation functions and requirements. The applicability exceptions of footnote (*) also apply to Functional Unit 5.a.				
b. Automatic Actuation Logic	2	1	2	1,2*,3*	20
c. Steam Generator Water level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1,2*,3*	19
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC)	3	2	3	1,2,3,4	13
7. UNDERVOLTAGE, VITAL BUS					
a. Loss of Voltage	1/bus	2	3	1,2,3	14
b. Sustained Degraded Voltage	3/bus	2/bus	3/bus	1,2,3	14

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. AUXILIARY FEEDWATER					
a. Automatic Actuation Logic **	2	1	2	1,2,3	20
b. NOT USED					
c. Steam Generator Water Level--Low-Low					
i. Start Motor Driven Pumps	3/stm. gen.	2/stm. gen. any stm.gen.	2/stm.gen.	1,2,3	19
ii.Start Turbine Driven Pumps	3/stm. gen.	2/stm. gen. any 2 stm.gen.	2/stm.gen.	1,2,3	19
d. Undervoltage - RCP Start Turbine - Driven Pump	4-1/bus	1/2 x 2	3	1,2	19
e. S.I. Start Motor-Driven Pumps	See 1 above (All S.I. initiating functions and requirements)				
f. Trip of Main Feedwater Pumps Start Motor Driven Pumps	2/pump	1/pump	1/pump	1,2	21
g. Station Blackout	See 6 and 7 above (SEC and U/V Vital Bus)				

TABLE 3.3-3 (Continued)

TABLE NOTATION

Trip function may be bypassed in this MODE below P-11.

Trip function may be bypassed in this MODE below P-12.

* Except when all main feedwater lines are isolated by (1) a closed and de-activated feedwater isolation valve, or (2) closed and de-activated feedwater regulating valve (FRV) and FRV bypass valves, or (3) a closed manual valve.

** Applies to Functional Unit 8 items c and d.

*** The automatic actuation logic includes two redundant solenoid operated vent valves for each Main Steam Isolation Valve (MSIV). Vent valves associated with an inoperable MSIV may be isolated provided that the MSIV is closed in accordance with actions of TS 3.7.1.5. One vent valve on any one of the remaining OPERABLE or open MSIVs may be isolated without affecting the function of the automatic actuation logic provided the remaining solenoid vent valves remain OPERABLE. The isolated MSIV vent valve shall be returned to OPERABLE status upon the first entry into MODE 5 following determination that the vent valve is inoperable. For any condition where more than one solenoid vent valve is inoperable for the OPERABLE or open MSIVs, entry into ACTION 20 is required.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 24 hours or, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.1 provided the other channel is OPERABLE.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST, provided the inoperable channel is placed in the tripped condition within 72 hours.

ACTION 15 - NOT USED

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated by CHANNEL CHECK within 72 hours; one additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 17 - With less than the Minimum Channels OPERABLE, operations may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-3 (Continued)

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours.

b. The Minimum Channels OPERABLE requirements is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.2.1.1.

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 24 hours or, be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may proceed provided that the inoperable channel is restored to OPERABLE within 72 hours.

ACTION 22 - NOT USED

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels ≥ 1925 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 3 of 4 T_{avg} channels at a setpoint of 543°F and T_{avg} increasing (with an allowable setpoint value ≤545°F)	P-12 prevents or defeats manual block of safety injection actuation high steam line flow and low steam line pressure.
	With 2 of 4 T_{avg} channels at a setpoint of 543°F and T_{avg} decreasing (with an allowable setpoint value ≥541°F)	Allows manual block of safety injection actuation on high steam line flow and low steam line pressure. Causes steam line isolation on high steam flow. Affects steam dump blocks.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	≤4.0 psig	≤4.5 psig
d. Pressurizer Pressure--Low	≥ 1765 psig	≥ 1755 psig
e. Differential Pressure Between Steam Lines--High	≤100 psi	≤112 psi
f. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low or Steam Line Pressure--Low	≤ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load Tavg ≥ 543°F ≥ 600 psig steam line pressure	≤ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load Tavg ≥ 541°F ≥ 579 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	≤ 15.0 psig	≤ 16.0 psig
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1. Manual	Not Applicable	Not Applicable
2. From Safety Injection Automatic Actuation Logic	Not Applicable	Not Applicable
b. Phase "B" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable
3. Containment Pressure--High-High	≤ 15.0 psig	≤ 16.0 psig
c. Containment Ventilation Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Containment Atmosphere Gaseous Radioactivity	Per Table 3.3-6	
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 15.0 psig	$\leq$ 16.0 psig
d. Steam Flow in Two Steam Lines-- High Coincident with Tavg -- Low-Low or Steam Line Pressure -- Low	$\leq$ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load. T avg $\geq$ 543°F $\geq$ 600 psig steam line pressure	$\leq$ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load. T avg $\geq$ 541°F $\geq$ 579 psig steam line pressure

TABLE 3.3-4 (continued)
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Safety Injection	Refer to Functional Unit 1 for all initiation functions and requirements.	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Steam Generator Water Level--High-High	≤ 67% of narrow range instrument span each steam generator	≤ 68% of narrow range instrument span each steam generator
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC)	Not Applicable	Not Applicable
7. UNDERVOLTAGE, VITAL BUS		
a. Loss of Voltage	≥ 70% of bus voltage	≥ 65% of bus voltage
b. Sustained Degraded Voltage	≥ 94.6% of bus voltage for ≤ 13 seconds	≥ 94% of bus voltage for ≤ 15 seconds
8. AUXILIARY FEEDWATER		
a. Automatic Actuation Logic	Not Applicable	Not Applicable
b. NOT USED		
c. Steam Generator Water Level--Low-Low	≥ 14.0% of narrow range instrument span each steam generator	≥ 13.0% of narrow range instrument span each steam generator
d. Undervoltage - RCP	≥ 70% RCP bus voltage	≥ 65% RCP bus voltage
e. S.I.	See 1 above (All S.I., setpoints)	
f. Trip of Main Feedwater Pumps	Not Applicable	Not Applicable
g. Station Blackout	See 6 and 7 above (SEC and Undervoltage, Vital Bus)	

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TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(7)	CHANNEL CALIBRATION(7)	CHANNEL FUNCTIONAL TEST(7)	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION				
a. Manual Initiation	N.A.	N.A.		1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	(2)	1,2,3,4
c. Containment Pressure--High			(3)	1,2,3
d. Pressurizer Pressure--Low				1,2,3
e. Differential Pressure Between Steam Lines--High				1,2,3
f. Steam Flow in Two Steam Lines--High coincident with Tavg--Low-Low or Steam Line Pressure-Low				1,2,3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.		1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	(2)	1,2,3,4
c. Containment Pressure--High-High			(3)	1,2,3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(7)	CHANNEL CALIBRATION(7)	CHANNEL FUNCTIONAL TEST(7)	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1. Manual	N.A.	N.A.		1,2,3,4
2. From Safety Injection Automatic Actuation Logic	N.A.	N.A.	(2)	1,2,3,4
b. Phase "B" Isolation				
1. Manual	N.A.	N.A.		1,2,3,4
2. Automatic Actuation Logic	N.A.	N.A.	(2)	1,2,3,4
3. Containment Pressure--High-High			(3)	1,2,3
c. Containment Ventilation Isolation				
1. Manual	N.A.	N.A.		1,2,3,4
2. Automatic Actuation Logic	N.A.	N.A.	(2)	1,2,3,4
3. Containment Atmosphere Gaseous Radioactivity - High	Per Surveillance Requirement 4.3.3.1			

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(7)	CHANNEL CALIBRATION(7)	CHANNEL FUNCTIONAL TEST(7)	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.		1,2(a),3**(a)
b. Automatic Actuation Logic	N.A.	N.A.	(2)	1,2(a),3(a)
c. Containment Pressure--High-High			(3)	1,2(a),3(a)
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low or Steam Line Pressure—Low				1,2(a),3(a)
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Safety Injection	Refer to Functional Unit 1 for all initiation functions and requirements. The applicability exceptions of footnote (b) also apply to Functional Unit 5.a.			
b. Automatic Actuation Logic	N.A.	N.A.	(2)	1,2(b),3(b)
c. Steam Generator Water Level--High-High				1,2(b),3(b)
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC) LOGIC				
a. Inputs	N.A.	N.A.	(6)	1,2,3,4
b. Logic, Timing and Outputs *	N.A.	N.A.	(1)	1,2,3,4
7. UNDERVOLTAGE, VITAL BUS				
a. Loss of Voltage				1,2,3
b. Sustained Degraded Voltage				1,2,3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(7)	CHANNEL CALIBRATION(7)	CHANNEL FUNCTIONAL TEST(7)	MODES IN WHICH SURVEILLANCE REQUIRED
8. AUXILIARY FEEDWATER				
a. Automatic Actuation Logic	N.A.	N.A.	(2)	1,2,3
b. NOT USED				
c. Steam Generator Water Level--Low-Low				1,2,3
d. Undervoltage - RCP				1,2
e. S.I.	See 1 above (All S.I. surveillance requirements)			
f. Trip of Main Feedwater Pumps	N.A.	N.A.		1,2
g. Station Blackout	See 6b and 7 above (SEC and U/V Vital Bus)			

TABLE 4.3-2 (Continued)

TABLE NOTATION

* Outputs are up to, but not including, the output relays.

** The provisions of Specification 4.0.4 are not applicable.

(1) Each logic channel shall be tested in accordance with the Surveillance Frequency Control Program. The CHANNEL FUNCTIONAL TEST of each logic channel shall verify that its associated diesel generator automatic load sequence timer is OPERABLE with the interval between each load block within 1 second of its design interval.

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

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

(4) NOT USED

(5) NOT USED

(6) Inputs from Undervoltage, Vital Bus, shall be tested in accordance with the Surveillance Frequency Control Program. Inputs from Solid State Protection System shall be tested in accordance with the Surveillance Frequency Control Program.

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

(a) Except when all MSIVs are closed.

(b) Except when all main feedwater lines are isolated by (1) a closed and de-activated feedwater isolation valve, or (2) closed and de-activated feedwater regulating valve (FRV) and FRV bypass valves, or (3) a closed manual valve.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, 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 shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations in accordance with the Surveillance Frequency Control Program.

TABLE 3.3-6
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Area	1	*	≤15 mR/hr	10^{-1}-10^{4} mR/hr	19
2. PROCESS MONITORS					
a. Containment					
1) Gaseous Activity					
a) Purge & Pressure - Vacuum Relief Isolation	1#	1,2,3,4&5	per ODCM Control 3.3.3.9	10^{1}-10^{6} cpm	23
b) RCS Leakage Detection	1	1,2,3&4	N/A	10^{1}-10^{6} cpm	20
2) Air Particulate Activity					
a) (NOT USED)					
b) RCS Leakage Detection	1	1,2,3&4	N/A	10^{1}-10^{6} cpm	20

* With fuel in the storage pool or building.

\# The plant vent noble gas monitor may also function in this capacity when the purge/pressure-vacuum relief isolation valves are open.

TABLE 3.3-6 (Continued)
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	≤ 3.0×10^{-2} µCi/cm^3 (Alarm only)	10^{-3}-10^{1} µCi/cm^3	23
2) High Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	≤ 1.0×10^{2} µCi/cm^3	10^{-1}-10^{5} µCi/cm^3 (Alarm only)	23
3) Condenser Exhaust System	1	1,2,3&4	≤ 1.27×10^{4} cpm (Alarm only)	1-10^{6} cpm	23
3. CONTROL ROOM					
a. Air Intake - Radiation Level	2/Intake##	**	≤ 2.48×10^{3} cpm	10^{1}-10^{7} cpm	24, 25

Control Room air intakes shared between Unit 1 and 2.

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 22 - (Not Used)

ACTION 23 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) 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 24 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status within 7 days or initiate and maintain operation of the Control Room Emergency Air Conditioning System (CREACS) in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

ACTION 25 - With no channels OPERABLE in a Control Room air intake, immediately initiate and maintain operation of the CREACS in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

TABLE 4.3-3

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INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, and CHANNEL CALIBRATION operations in accordance with the Surveillance Frequency Control Program.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Pressurizer Pressure	Hot Shutdown Panel 213	1700-2500 psig	1
2. Pressurizer Level	Hot Shutdown Panel 213	0 - 100%	1
3. Steam Generator Pressure	Hot Shutdown Panel 213	0 - 1200 psig	1/steam generator
4. Steam Generator Level	Hot Shutdown Panel 213	0 - 100%	1/steam generator

TABLE 4.3-6

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

---NOTE---
Separate Condition entry is allowed for each Function.
----

a. As shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-11.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1	1, 2
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1	1, 2
3. Reactor Coolant Pressure (Wide Range)	2	1	1, 2
4. Pressurizer Water Level	2	1	1, 2
5. Steam Line Pressure	2/Steam Generator	1/Steam Generator	1, 2
6. Steam Generator Water Level (Narrow Range)	2/Steam Generator	1/Steam Generator	1, 2
7. Steam Generator Water Level (Wide Range)	4 (1/Steam Generator)	3 (1/Steam Generator)	1, 2
8. Refueling Water Storage Tank Water Level	2	1	1, 2
9. deleted			
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	3 (1/Steam Generator)	4, 6
11. Deleted			
12. Deleted			

TABLE 3.3-11 (CONTINUED)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
13. Deleted			
14. Deleted			
15. Deleted			
16. Containment Pressure - Wide Range	2	1	7, 2
17. Containment Water Level - Wide Range	2	1	7, 2
18. Core Exit Thermocouples	4/core quadrant	2/core quadrant	1, 2
19. Reactor Vessel Level Instrumentation System (RVLIS)	2	1	8, 9
20. Containment High Range Accident Radiation Monitor	2	2	10
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor	1/MS Line	1/MS Line	10
22. Wide Range Neutron Flux Monitors	2	1	1, 2
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level	2	1	1, 2
24. Containment Isolation Valve Position Indication	2 per penetration flow path [a][b]	1/valve[c]	1, 2

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

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

(c) Action 2 not required for penetration flow paths with only one installed control room indication channel.

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 1	With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.
ACTION 2	With the number of OPERABLE accident monitoring channels less than the MINIMUM Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
ACTION 3	deleted
ACTION 4	With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate channel; otherwise, restore the inoperable channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.
ACTION 5	deleted

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 6 With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 7 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed until the next CHANNEL CALIBRATION (which shall be performed upon the next entry into MODE 5, COLD SHUTDOWN).

ACTION 8 With one RVLIS channel inoperable, restore the RVLIS channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 9 With both RVLIS channels inoperable, restore one channel to OPERABLE status within 7 days or submit a special report in accordance with Specification 6.9.4.

ACTION 10 With the number of OPERABLE Channels less than required by the minimum channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

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

TABLE 4.3-11
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK[(1)]	CHANNEL CALIBRATION[(1)]	CHANNEL FUNCTIONAL TEST[(1)]
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)			N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)			N.A.
3. Reactor Coolant Pressure (Wide Range)			N.A.
4. Pressurizer Water Level			N.A.
5. Steam Line Pressure			N.A.
6. Steam Generator Water Level (Narrow Range)			N.A.
7. Steam Generator Water Level (Wide Range)			N.A.
8. Refueling Water Storage Tank Water Level			N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	S/U#		N.A.
11. Deleted			

Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK(1)	CHANNEL CALIBRATION(1)	CHANNEL FUNCTIONAL TEST(1)
12. Deleted			
13. Deleted			
14. Deleted			
15. Deleted			
16. Containment Pressure - Wide Range			N.A.
17. Containment Water Level - Wide Range			N.A.
18. Core Exit Thermocouples			N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)			N.A.
20. Containment High Range Accident Radiation Monitor			
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor			
22. Wide Range Neutron Flux Monitors			N.A.
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level			N.A.
24. Containment Isolation Valve Position Indication			N.A.

Table Notation

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

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be operable to ensure that the limits of ODCM Control 3.11.1.1 are not exceeded.

APPLICABILITY: At all times.

ACTION:

a. Not Used

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Exert best efforts to return the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next annual radioactive effluent release report why the inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-12.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Not Used		
2. Not Used		
3. Not Used		
4. TANK LEVEL INDICATING DEVICES		
a. Temporary Outsi·'e Storage Tanks as Required	1	30

TABLE NOTATION

ACTION 26 - Not Used

ACTION 27 - Not Used

ACTION 28 - Not Used

ACTION 29 - Not Used

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue for up to 30 days provided the tank liquid level is estimated during all liquid additions to the tank.

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK(1)	SOURCE CHECK(1)	CHANNEL CALIBRATION(1)	CHANNEL FUNCTIONAL TEST(1)
1. Not Used				
2. Not Used				
3. Not Used				
4. TANK LEVEL INDICATING DEVICES**				
a. Temporary Outside Storage Tanks as Required	D*	N.A.		

TABLE NOTATION

* During liquid additions to the tank.

** If tank level indication is not provided, verification will be done by visual inspection.

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

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POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.14 The Power Distribution Monitoring System (PDMS) shall be OPERABLE with:

a. A minimum of the following inputs from the plant available for use by the PDMS as defined in Table 3.3-14:

   1. Control Bank Position

   2. T_{cold}

   3. Reactor Power Level

   4. NIS Power Range Detector Section Signals

b. Core Exit Thermocouples (T/C) meeting the criteria:

   1. At least 25% operable T/C with at least 2 T/C per quadrant, and

   2. The T/C pattern has coverage of all interior fuel assemblies (no face along the baffle), within a chess knight's move, radially, from a responding, calibrated T/C, or

   3. At least 25%, operable T/C with at least 2 T/C per quadrant, and the installed PDMS calibration was determined within the last 31 Effective Full Power Days (EFPD).

   4. The T/C temperatures used by the PDMS are calibrated via cross calibration with the loop temperature measurement RTDs, and using the T/C flow mixing factors determined during installed PDMS calibration.

c. An installed PDMS calibration satisfying the criteria:

   1. The initial calibration in each operating cycle is determined using measurements from at least 75% of the incore movable detector thimbles obtained at a THERMAL POWER greater than 25% of RATED THERMAL POWER.

   2. The calibration is determined using measurements from at least 50% of the incore movable detector thimbles at any time except as specified in 3.3.3.14.c.1, and

   3. The calibration is determined using a minimum of 2 detector thimbles per core quadrant.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILTY: MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.14.1 The operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, and 3.3.3.14.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.14.2 Calibration of the PDMS is required:

a. In accordance with the Surveillance Frequency Control Program when the minimum number and core coverage criteria as defined in 3.3.3.14.b.1 and 3.3.3.14.b.2 are satisfied, or

b. In accordance with the Surveillance Frequency Control Program when only the minimum number criterion as defined in 3.3.3.14.b.3 is satisfied.

TABLE 3.3-14

REQUIRED PDMS PLANT INPUT INFORMATION

PLANT INPUT INFORMATION	AVAILABLE INPUTS	MINIMUM NO. OF VALID INPUTS	APPLICABLE MODES
Control Bank Position	4	4[a]	1[c]
T_{cold}	4	2	1[c]
Reactor Power Level	3	1[b]	1[c]
NIS Power Range Excore Detector Section Signals	8	6[d]	1[c]

TABLE NOTATIONS

a. Determined from either valid Demand Position or the average of the valid individual RCCA position indications for all RCCAs in the Control Bank.

b. Determined from either the reactor THERMAL POWER derived using a valid secondary calorimetric measurement, the average NIS Power Range Detector Power, or the average RCS Loop ΔT.

c. Greater than 25% RTP.

d. Comprised of an upper and lower detector section signal per Power Range Channel; a minimum of 3 OPERABLE channels required.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS

NORMAL OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENT

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

* See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 11 and its associated steam generator and reactor coolant pump,

2. Reactor Coolant Loop 12 and its associated steam generator and reactor coolant pump,

3. Reactor Coolant Loop 13 and its associated steam generator and reactor coolant pump,

4. Reactor Coolant Loop 14 and its associated steam generator and reactor coolant pump.

b. At least one of the above coolant loops shall be in operation* when the rod control system is deenergized**.

c. All of the above coolant loops shall be in operation when the rod control system is energized**.

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

REACTOR COOLANT SYSTEM

HOT STANDBY

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 5% (narrow range) in accordance with the Surveillance Frequency Control Program.

---

* All reactor coolant pumps may be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration (2) core outlet temperature is maintained at least 10°F below saturation temperature, and (3) the rod control system is de-energized**

** The rod control system shall be considered de-energized when one or more of the following conditions exist:

1) Both Rod Drive MG set motor breakers are open.

2) Both Rod Drive MG set generator breakers are open.

3) A combination of at least three of the Reactor Trip and/or Reactor Trip Bypass Breakers are open.

If none of the above conditions for de-energizing the rod control system are met; the system shall be considered energized.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (11) and its associated steam generator and reactor coolant pump,*

2. Reactor Coolant Loop (12) and its associated steam generator and reactor coolant pump,*

3. Reactor Coolant Loop (13) and its associated steam generator and reactor coolant pump,*

4. Reactor Coolant Loop (14) and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal Loop (11),

6. Residual Heat Removal Loop (12).

b. At least one of the above coolant loops shall be in operation.**

APPLICABILITY: MODE 4

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to the POPS enable temperature specified in the PTLR unless 1) the pressurizer water volume is less than 1650 cubic feet (93.2% of pressurizer level indication) or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per the inservice testing schedule.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to 5% of narrow range in accordance with the Surveillance Frequency Control Program.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# residual heat removal loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5.##

ACTION:

a. With less than the above required loops operable, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

One RHR loop may be inoperable for up to two hours for surveillance testing, provided the other RHR loop is OPERABLE and in operation. Additionally, four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop.

A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to the POPS enable temperature specified in the PTLR unless 1) the pressurizer water volume is less than 1650 cubic feet (equivalent to approximately 93.2% of level), or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

** The residual heat removal pumps may be de-energized for up to 2 hours provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

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REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE* with a lift setting of 2485 psig ± 3%.**,***

APPLICABILITY: MODE 4 and 5

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

* While in Mode 5, an equivalent size vent pathway may be used provided that the vent pathway is not isolated or sealed.

** The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

*** Following testing the lift setting shall be reset to within ± 1%.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 3%.*,**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes, or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

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* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Following testing the lift setting shall be reset to within ± 1%.

REACTOR COOLANT SYSTEM

3/4.4.3 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3 Two power relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 6 hours either restore at least one PORV to OPERABLE status or close the associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Restore the remaining PORV to OPERABLE status within 72 hours from failure of that valve to meet the Limiting Condition for Operation.

d. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place the associated PORV in manual control; restore the block valve to operable status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With both block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place the associated PORVs in manual control; restore at least one block valve to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Restore the remaining block valve to OPERABLE status within 72 hours from failure of that valve to meet the Limiting Condition for Operation.

SURVEILLANCE REQUIREMENTS

4.4.3.1 In addition to the requirements of the INSERVICE TESTING PROGRAM, each PORV shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

b. Operating solenoid valves, air control valves, and check valves on associated air accumulators in PORV control systems through one complete cycle of full travel, and

c. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.3.2 Each block valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b, or c in Specification 3.4.3.

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1650 cubic feet (92% indicated level), and at least two groups of pressurizer heaters each having a capacity of ≥ 150 kw and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

4.4.4.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring circuit current in accordance with the Surveillance Frequency Control Program.

4.4.4.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by manually transferring power from the normal to the emergency power supply and energizing the heaters.

STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained and all SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a.* With one or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program:

1. Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days; and

2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

b. With SG tube integrity not maintained or the required Action of a. above not met, be in at least HOT STANDBY within 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection.

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* Separate Action is allowed for each SG tube.

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REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level monitoring system, and

c. Either the containment fan cooler condensate flow rate or the containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring systems-performance of CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

b. Containment sump level and containment fan cooler condensate flow rate (if being used) monitoring systems-performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and primary-to-secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by;

a. Monitoring the containment atmosphere particulate radioactivity monitor in accordance with the Surveillance Frequency Control Program.

b. Monitoring the containment sump inventory in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (Continued)

c.* Verifying primary-to-secondary leakage is ≤ 150 gallons per day through any one steam generator in accordance with the Surveillance Frequency Control Program during steady state operation.

d.* Performance of a Reactor Coolant System water inventory balance** in accordance with the Surveillance Frequency Control Program. The water inventory balance shall be performed with the plant at steady state conditions. The provisions of specification 4.0.4 are not applicable for entry into Mode 4, and

e. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program.

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* Not required to be completed until 12 hours after establishment of steady state operation.

** Not applicable to primary-to-secondary leakage.

REACTOR COOLANT SYSTEM

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor Coolant System Pressure Isolation Valves shall be OPERABLE.

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

ACTION:

a. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the specified limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.3 Each Reactor Coolant System Pressure Isolation Valve shall be demonstrated OPERABLE pursuant to the INSERVICE TESTING PROGRAM, except that in lieu of any leakage testing required by the INSERVICE TESTING PROGRAM, each valve shall be demonstrated OPERABLE by verifying leakage to be ≤ 5.0 gpm each valve[(a)(b)]:

a. In accordance with the Surveillance Frequency Control Program.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

d. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valve. For all other systems testing will be done once per refueling.

The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or 4.

SURVEILLANCE REQUIREMENTS (Continued)

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable. However, for initial tests, or tests following valve repair or replacement, leakage rates less than or equal to 5.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum differential test pressure shall not be less than 150 psid.

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REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. ≤ 600 μCi/gm DOSE EQUIVALENT XE-133.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

NOTE

Specification 3.0.4.c is applicable

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a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131:

1. Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gram at least once every 4 hours and restore DOSE EQUIVALENT I-131 to ≤ 1.0 μCi/gram within 48 hours, or

2. Be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the specific activity of the primary coolant > 600 μCi/gram DOSE EQUIVALENT XE-133:

1. Restore DOSE EQUIVALENT XE-133 to ≤ 600 μCi/gram within 48 hours, or

2. Be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

NOTE

SR 4.4.8.1 is not required to be performed in MODE 4, and is not required to be performed in MODE 3 until 24 hours after $T_{avg} \geq 500^{o}F$.

4.4.8.1 Verify the specific activity of the primary coolant ≤600 μCi/gm DOSE EQUIVALENT XE-133 in accordance with the Surveillance Frequency Control Program.

4.4.8.2 Verify the specific activity of the primary coolant ≤1.0 μCi/gm DOSE EQUIVALENT I-131 in accordance with the Surveillance Frequency Control Program, and between 2 and 6 hours after a THERMAL POWER change of ≥15% RATED THERMAL POWER within a one hour period.

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limits specified in the PTLR with:

a. A maximum heatup rate within the limits specified in the PTLR,

b. A maximum cooldown rate within the limits specified in the PTLR, and

c. A maximum temperature change within limits specified in the PTLR during hydrostatic testing operations above system design pressure.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits specified in the PTLR in accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update the P-T Limit Curves specified in the PTLR.

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PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program during auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two Pressurizer Overpressure Protection System relief valves (POPS) with a lift setting of less than or equal to the value specified in the PTLR, or

b. A reactor coolant system vent of greater than or equal to 3.14 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, except when the reactor vessel head is removed.

ACTION:

a. With one POPS inoperable in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, either restore the inoperable POPS to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

b. With one POPS inoperable in MODES 5 or 6 with the Reactor Vessel Head installed, restore the inoperable POPS to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

c. With both POPSs inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

d. In the event either the POPS or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the POPS or vent(s) on the transient and any corrective action necessary to prevent recurrence.

e. LCO 3.0.4.b is not applicable when entering MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each POPS shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the POPS actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the POPS is required OPERABLE, and in accordance with the Surveillance Frequency Control Program thereafter when the POPS is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel in accordance with the Surveillance Frequency Control Program.

c. Verifying the POPS isolation valve is open in accordance with the Surveillance Frequency Control Program when the POPS is being used for overpressure protection.

d. Testing pursuant to the INSERVICE TESTING PROGRAM.

4.4.9.3.2 The RCS vent(s) shall be verified to be open in accordance with the Surveillance Frequency Control Program* when the vents(s) is being used for overpressure protection.

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* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open in accordance with the Surveillance Frequency Control Program.

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SECTION 3/4.4.11
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This page is effective as of its date of issuance (8-28-89) and is to be implemented prior to Reactor startup following the next plant shutdown to Mode 3, Hot Standby.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained volume of between 6,223 and 6,500 gallons of borated water,

c. A boron concentration of between 2,200 and 2,500 ppm, and,

d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration outside the required limits, restore the inoperable accumulator to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within 24 hours and be in HOT SHUTDOWN within the next 12 hours.

c. With the boron concentration of one accumulator outside the required limits, restore the boron concentration to within the required limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than or equal to 1000 psig within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

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

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

* Pressurizer Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of ≥ 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. In accordance with the Surveillance Frequency Control Program when the RCS pressure is greater than 1000 psig by verifying that the power lockout switch is in lockout.

d. In accordance with the Surveillance Frequency Control Program by verifying that each accumulator isolation valve opens automatically upon receipt of a safety injection test signal.

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of the following injection systems:

a. One OPERABLE centrifugal charging pump and associated flow path capable of taking suction from the refueling water storage tank and transferring suction to the residual heat removal pump discharge piping and;

   1. Discharging into each Reactor Coolant System (RCS) cold leg.

b. One OPERABLE safety injection pump and associated flow path capable of taking suction from the refueling water storage tank and transferring suction to the residual heat removal pump discharge piping and;

   1. Discharging into each RCS cold leg, and; upon manual initiation,

   2. Discharging into its two associated RCS hot legs.

c. One OPERABLE residual heat removal pump and associated residual heat removal heat exchanger and flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation and;

   1. Discharging into each RCS cold leg.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS 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.

c. With both ECCS subsystems inoperable for surveillance testing, restore at least one subsystem to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours and at least COLD SHUTDOWN within the subsequent 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

1. Verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. 1 SJ 69	a. RHR pump suction	a. open
b. 1 SJ 30	b. SI pump suction	b. open
c. 11 SJ 40	c. SI discharge to hot legs	c. closed
d. 12 SJ 40	d. SI discharge to hot legs	d. closed
e. 1 RH 26	e. RHR discharge to hot legs	e. closed
f. 11 SJ 49	f. RHR discharge to cold legs	f. open
g. 12 SJ 49	g. RHR discharge to cold legs	g. open
h. 1 CS 14#	h. Spray additive tank discharge	h. open
i. 1 SJ 135	i. SI discharge to cold legs	i. open
j. 1 SJ 67	j. SI recirc. line isolation	j. open
k. 1 SJ 68	k. SI recirc. line isolation	k. open
l. 11 SJ 44	l. Containment sump isolation valve	l. closed
m. 12 SJ 44	m. Containment sump isolation valve	m. closed

2. Verifying that the following valves are in the indicated positions:

Valve Number	Valve Function	Valve Position
a. 11 RH 19	a. RHR crosstie valve	a. open
b. 12 RH 19	b. RHR crosstie valve	b. open

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

1. 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.

2. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points.

\# If inoperable, the applicable Technical Specification is 3.6.2.2.

SURVEILLANCE REQUIREMENTS (Continued)

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. In accordance with the Surveillance Frequency Control Program the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

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

   1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. In accordance with the Surveillance Frequency Control Program, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

SURVEILLANCE REQUIREMENTS (Continued)

f. By verifying that each of the following pumps develops the indicated Total Dynamic Head (TDH) when tested at the test flow point pursuant to the INSERVICE TESTING PROGRAM:

1. Centrifugal charging pump ≥ 2338 psi TDH
2. Safety Injection Pump ≥ 1369 psi TDH
3. Residual heat removal pump ≥ 165 psi TDH

g. By verifying the correct position of each of the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.
2. In accordance with the Surveillance Frequency Control Program.

HPSI SYSTEM VALVE NUMBER	LPSI SYSTEM VALVE NUMBER
11 SJ 16	11 SJ 138
12 SJ 16	12 SJ 138
13 SJ 16	13 SJ 138
14 SJ 16	14 SJ 138
	11 SJ 143
	12 SJ 143
	13 SJ 143
	14 SJ 143

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1. For Safety Injection pumps, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is ≥ 453 gpm; and

b) The total flow rate through all four injection lines is ≤ 647 gpm, and

c) The difference between any pair of injection line flow rates is ≤ 12.0 gpm, and

d) The total pump flow rate is ≤ 664 gpm in the cold leg alignment, and

e) The total pump flow rate is ≤ 654 gpm in the hot leg alignment.

SURVEILLANCE REQUIREMENTS (Continued)

2. For Centrifugal Charging pumps, with a single pump running:

   a) The sum of the injection line flow rates, excluding the highest flow rate, is ≥ 306 gpm, and

   b) The total flow rate through all four injection lines is ≤ 444 gpm, and

   c) The difference between any pair of injection line flow rates is ≤ 10.5 gpm, and

   d) The total pump flow rate is ≤ 554 gpm.

i. The automatic interlock function of the RHR System shall be verified within the seven (7) days prior to placing the RHR System in service for cooling of the Reactor Coolant System. This shall be done by verifying with a test signal corresponding to a reactor coolant pressure of 375 psig or greater, that the 1RH1 and 1RH2 valves cannot be opened.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - Tavg < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump# and associated flow path capable of taking suction from the refueling water storage tank and transferring suction to the residual heat removal pump discharge piping and;

   1. Discharging into each Reactor Coolant System (RCS) cold leg.

b. One OPERABLE residual heat removal pump and associated residual heat removal heat exchanger and flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation and;

   1. Discharging into each RCS cold leg, and; upon manual initiation,

   2. Discharging into two RCS hot legs.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS 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.

d. LCO 3.0.4.b is not applicable to ECCS high head subsystem

A maximum of one safety injection pump or one centrifugal charging pump shall be OPERABLE in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, Mode 5, or Mode 6 when the head is on the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All safety injection pumps and centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated to be inoperable in accordance with the Surveillance Frequency Control Program while in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, MODE 5, or MODE 6 when the head is on the reactor vessel by either of the following methods:

a. By verifying that the motor circuit breakers have been removed from their electrical power supply circuits or,

b. For verifying that the pump is in a recirculation flow path and that two independent means of preventing RCS injection are utilized.

EMERGENCY CORE COOLING SYSTEMS

SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

3.5.4 Reactor coolant pump seal injection flow shall be ≤ 40 gpm with centrifugal charging pump discharge header pressure ≥ 2430 psig and the charging flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With seal injection flow not within the limit, adjust manual seal injection throttle valves to give a flow within the limit with the charging pump discharge pressure ≥ 2430 psig and the charging flow control valve full open within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 In accordance with the Surveillance Frequency Control Program, verify manual seal injection throttle valves are adjusted to give a flow within the limit with centrifugal charging pump discharge header pressure ≥ 2430 psig, and the charging flow control valve full open.

The provisions of Specification 4.0.4 are not applicable for entry into Mode 3. This exemption is allowed for up to 4 hours after the Reactor Coolant System pressure stabilizes at 2235 ± 20 psig.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained volume of $\geq$ 364,500 gallons of borated water.

b. A boron concentration of between 2300 and 2500 ppm, and

c. RWST borated water temperature $\geq$ 35°F and $\leq$ 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the RWST inoperable due to boron concentration or temperature not within limits, restore the tank to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the RWST inoperable for reasons other than boron concentration or temperature not within limits, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

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

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. In accordance with the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is < 35°F or > 100°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a1. In accordance with the Surveillance Frequency Control Program by verifying that each containment manual valve or blind flange that is located outside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. Valves and blind flanges in high radiation areas may be verified by use of administrative controls.

a2. Prior to entering Mode 4 from Mode 5 if not performed within the last 92 days by verifying that each containment manual valve or blind flange that is located inside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. Valves and blind flanges in high radiation areas may be verified by use of administrative controls.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. In accordance with the Surveillance Frequency Control Program by verifying that the surveillance requirements of 4.6.2.3.a are met for penetrations associated with the containment fan coil units.

d. In accordance with the Surveillance Frequency Control Program by verifying that the surveillance requirements of 4.6.2.3.d are met for penetrations associated with the containment fan coil units.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate (Type A test) in accordance with the Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Containment Leakage Rate Testing Program for all penetrations and valves subject to Type B and C tests.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage (Type A test) not in accordance with the Containment Leakage Rate Testing Program, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests not in accordance with the Containment Leakage Rate Testing Program, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated as follows:

a. Type A tests shall be in accordance with the Containment Leakage Rate Testing Program.

b. Type B and C tests shall be conducted in accordance with the Containment Leakage Rate Testing Program.

c. Air locks shall be tested and demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each 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 Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

Notes

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

(2) Separate condition entry is allowed for each air lock.

(3) Required ACTIONS a.1, a.2, and a.3 are not applicable if both doors in the same air lock are inoperable and condition c. is entered.

(4) Required ACTIONS b.1, b.2, and b.3 are not applicable if both doors in the same air lock are inoperable and condition c. is entered.

(5) Enter applicable Conditions and required Actions of LCO 3.6.1, "Primary Containment," when air lock leakage results in exceeding the overall containment leakage rate.

a. One or more containment air locks with one containment airlock door inoperable:

1. Verify the OPERABLE door is closed in the affected air lock within 1 hour, and:

2. Lock the OPERABLE door closed in the affected air lock within 24 hours, and:

3. Verify the OPERABLE door is locked closed in the affected air lock once per 31 days. Entry and exit is permissible for 7 days (from initial LCO entry) under administrative controls if one door is inoperable in each air lock. Air lock doors in high radiation areas may be verified locked closed by administrative means.

b. One or more containment air locks with only the containment air lock interlock mechanism inoperable.

1. Verify an OPERABLE door is closed in the affected air lock within 1 hour, and:

2. Lock an OPERABLE door closed in the affected air lock within 24 hours, and:

3. Verify an OPERABLE door is locked closed in the affected air lock once per 31 days. Entry and exit of containment is permissible under the control of a dedicated individual for the duration of the entry to ensure only one door is open at a time. Air lock doors in high radiation areas may be verified locked closed by administrative means.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATIONS (Continued)

c. One or more containment air locks inoperable for reasons other than condition a. or b.

   1. Immediately initiate action to evaluate overall containment leakage per LCO 3.6.1, and:

   2. Verify that at least one door is closed in the affected air lock within 1 hour, and:

   3. Restore the air lock to OPERABLE status within 24 hours.

d. If the ACTIONS and associated completion times of a., b., or c. cannot be met, be in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying seal leakage rate in accordance with the Containment Leakage Rate Testing program.

b. By conducting an overall air lock leakage test in accordance with the 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.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -1.5 and +0.3 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 Verify the Containment Average Air Temperature is within limit in accordance with the Surveillance Frequency Control Program.

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6. 1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined in accordance with the Containment Leakage Rate Testing Program.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be evaluated for reportability pursuant to 10CFR50.72 and 10CFR50.73. The evaluation shall be documented and shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective action taken.

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Note that the elements of TS 3.6.1.7 and 4.6.1.7 were relocated to TS 3/4.6.3.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and transferring suction to the RHR pump discharge.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray 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 on recirculation flow, each pump develops a differential pressure of greater than or equal to 204 psid when tested pursuant to the Inservice Testing Program.

c. In accordance with the Surveillance Frequency Control Program during shutdown, by:

    1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment High-High pressure test signal.

    2. Verifying that each spray pump starts automatically on a Containment High-High pressure test signal.

d. Following activities that could result in nozzle blockage, either evaluate the work performed to determine the impact to the containment spray system, or perform an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and transferring suction to the RHR pump discharge.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray 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 on recirculation flow, each pump develops a differential pressure of greater than or equal to 204 psid when tested pursuant to the INSERVICE TESTING PROGRAM.

c. In accordance with the Surveillance Frequency Control Program during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment High-High pressure test signal.

   2. Verifying that each spray pump starts automatically on a Containment High-High pressure test signal.

d. Following activities that could result in nozzle blockage, either evaluate the work performed to determine the impact to the containment spray system, or perform an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SPRAY ADDITIVE SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the spray additive tank eductor flow will be 35 ± 3.5 gpm to each containment spray system. Testing may be performed by measuring the flow of borated water from the RWST through the installed 2" test line and Valve CS31; using this test line up with the spray pump operating in the recirculation mode and the RWST level at 41 feet ± 0.5 feet, the measured flow shall be 57 gpm ± 5.7 gpm.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Five containment cooling fans shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one or two of the above required containment cooling fans inoperable, restore the inoperable cooling fan(s) to OPERABLE status within 14 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With three or more of the above required containment cooling fans inoperable, restore at least three cooling fans to OPERABLE status within 1 hour or be in at least HOT STANDBY WITHIN the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the remaining inoperable cooling fans to OPERABLE status within 14 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling fan shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

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

   1. Verifying the water level in each service water accumulator vessel is greater than or equal to 226 inches and less than or equal to 252 inches.

   2. Verifying the temperature in each service water accumulator vessel is greater than or equal to 55°F and less than or equal to 95°F.

   3. Verifying the nitrogen cover pressure in each service water accumulator vessel is greater than or equal to 135 psig and less than or equal to 160 psig.

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

   1. Starting (unless already operating) each fan from the control room in low speed.

   2. Verifying that each fan operates for at least 15 minutes in low speed.

   3. Verifying a cooling water flow rate of greater than or equal to 1300 gpm to each cooler.

c. In accordance with the Surveillance Frequency Control Program by verifying that on a safety injection test signal:

   1. Each fan starts automatically in low speed.

   2. The automatic valves and dampers actuate to their correct positions and that the cooling water flow rate to each cooler is greater than or equal to 1300 gpm.

d. In accordance with the Surveillance Frequency Control Program by verifying that on a loss of offsite power test signal, each service water accumulator vessel discharge valve response time is within limits.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

NOTE 1

Penetration flow paths, except for the containment purge valves, may be unisolated intermittently under administrative controls.

Note 2

A containment purge valve is not a required containment isolation valve when its flow path is isolated with a testable blind flange tested in accordance with SR 4.6.1.2.b. The *required* containment purge supply and exhaust isolation valves shall be closed. (Valves immobilized in shut position with control air to valve operators isolated and tagged out of service).

NOTE 3

The containment pressure-vacuum relief isolation valves may be opened on an intermittent basis, under administrative control, as necessary to satisfy the requirement of Specification 3.6.1.4.

1. With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

   a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

   c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

   d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one required containment purge supply and/or exhaust isolation valve open, close the open valve(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 DELETED

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Not used.

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each required Purge and each Pressure-Vacuum Relief valve actuates to its isolation position.

e. Verifying that the Containment Pressure-Vacuum Relief Isolation valves are limited to ≤ 60% opening angle.

4.6.3.1.3 In accordance with the Surveillance Frequency Control Program, verify that on a main steam isolation test signal, each main steam isolation valve actuates to its isolation position.

4.6.3.1.4 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

4.6.3.1.5 Each required containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then in accordance with the Surveillance Frequency Control Program, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2.b for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60La.

4.6.3.1.6 A pressure drop test to identify excessive degradation of resilient valve seals shall be conducted on the:

a. Required Containment Purge Supply and Exhaust Isolation Valves in accordance with the Surveillance Frequency Control Program.

b. Deleted.

4.6.3.1.7 The required containment purge supply and exhaust isolation valves shall be determined closed in accordance with the Surveillance Frequency Control Program.

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves (MSSVs) associated with each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or two main steam line code safety valves inoperable in one or more steam generators, operation in Modes 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or reduce power to less than or equal to the applicable percent of RATED THERMAL POWER per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With three main steam line code safety valves inoperable in one or more steam generators, operation in Modes 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valves are restored to OPERABLE status or reduce power to less than or equal to the applicable percent of RATED THERMAL POWER per Table 3.7-1 and within 36 hours, reduce the Power Range Neutron Flux High trip setpoint to less than or equal to the RATED THERMAL POWER per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Verify each required MSSV lift setpoint per Table 4.7-1. No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER
WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power* (Percent of RATED THERMAL POWER)
1	87
2	59
3	39

*The values do not provide any allowance for calorimetric error.

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TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	Loop A	Loop B	Loop C	Loop D	LIFT SETTING (± 3%)*	ORIFICE SIZE (sq. inches)
a.	11MS11	12MS11	13MS11	14MS11	1125 psig	16.0
b.	11MS12	12MS12	13MS12	14MS12	1120 psig	16.0
c.	11MS13	12MS13	13MS13	14MS13	1110 psig	16.0
d.	11MS14	12MS14	13MS14	14MS14	1100 psig	16.0
e.	11MS15	12MS15	13MS15	14MS15	1070 psig	16.0

*Following testing the lift setting shall be reset to within ± 1%.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated manual activation switches in the control room and flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate vital busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective ACTION to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

d. LCO 3.0.4.b is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

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

   1. Verifying that each non-automatic valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

   2. Verify the manual maintenance valves in the flow path to each steam generator are locked open.

SURVEILLANCE REQUIREMENTS (continued)

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

1. Verify that the developed head of each motor driven pump at the flow test point is greater than or equal to the required developed head.

2. Verify that the developed head of the steam driven pump at the flow test point is greater than or equal to the required developed head when the steam generator pressure is > 680 psig. The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 24 hours after secondary side pressure is greater than 680 psig.

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

1. Verifying that each auxiliary feedwater automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

2. Verifying that each auxiliary feedwater pump starts automatically on an actual or simulated actuation signal.

The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

PLANT SYSTEMS

AUXILIARY FEED STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The auxiliary feed storage tank (AFST) shall be OPERABLE with a minimum contained volume of 200,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the auxiliary feed storage tank inoperable, within 4 hours either:

a. Restore the AFST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of a demineralized water or a fire protection/domestic water storage tank as a backup supply to the auxiliary feedwater pumps and restore the auxiliary feed storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The auxiliary feed storage tank shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the water level is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 A demineralized water storage tank shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the tank contains ≥ 200,000 gallons of water and by verifying proper alignment of valves for taking suction from this tank when it is the supply source for the auxiliary feedwater pumps.

4.7.1.3.3 A fire protection/domestic water storage tank shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the tank contains ≥ 200,000 gallons of water and by verifying proper alignment of valves for taking suction from this tank when it is the supply source for the auxiliary feedwater pumps.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$0.10 μCi/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	In accordance with the Surveillance Frequency Control Program
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODE 1
MODES 2 and 3 except when all MSIVs are closed

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

otherwise, be in MODE 2 within the next 6 hours.

MODES 2 - and 3 With one or more main steam line isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may proceed provided;

a. The isolation valve(s) is (are) maintained closed, and

b. The isolation valve(s) is (are) verified closed once per 7 days.

Otherwise, be in MODE 3, HOT STANDBY, within the next 6 hours, and MODE 4, HOT SHUTDOWN, within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable.

LIMITING CONDITION FOR OPERATION

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generator shall be determined to be < 200 psig in accordance with the Surveillance Frequency Control Program when the temperature of either the primary or secondary coolant is < 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE 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.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours * or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two service water 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 during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on Safeguards Initiation signal.

* Operation with only the 11 service water loop OPERABLE may continue for up to 10 days. This note is applicable for one time use during Salem Unit No. 1 Cycle 15.

3/4.7.5 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.5.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Delaware River exceeds 10.5' Mean Sea Level USGS datum, at the service water intake structure.

APPLICABILITY: At all times.

ACTION:

a. With the water level at the service water intake structure above elevation 10.5' Mean Sea Level USGS datum, close all watertight doors within 2 hours.

b. With the water level at the service water intake structure above elevation 11.5' Mean Sea Level USGS datum, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The water level at the service water intake structure shall be determined to be within the limits by:

a. Measurement in accordance with the Surveillance Frequency Control Program when the water level is below elevation 10.5' Mean Sea Level USGS datum, and

b. Measurement in accordance with the Surveillance Frequency Control Program when the water level is equal to or above elevation 10.5' Mean Sea Level USGS datum.

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The common control room emergency air conditioning system (CREACS)* shall be OPERABLE with:

a. Two independent air conditioning filtration trains (one from each unit) consisting of:

   1. Two fans and associated outlet dampers,
   2. One cooling coil,
   3. One charcoal adsorber and HEPA filter array,
   4. Return air isolation damper.

b. All other automatic dampers required for operation in the pressurization or recirculation modes.

c. The control room envelope intact.

NOTE: The control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: ALL MODES and during movement of irradiated fuel assemblies.

ACTION: MODES 1, 2, 3, and 4

a. With one filtration train inoperable, align CREACS for single filtration train operation** within 4 hours, and restore the inoperable filtration train to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With CREACS aligned for single filtration train operation and with one of the two remaining fans or associated outlet damper inoperable, restore the inoperable fan or damper to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the Control Room Envelope boundary inoperable:

   1. Immediately, initiate action to implement mitigating actions, and

   2. Within 24 hours, verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, and

   3. Within 90 days, restore the Control Room Envelope boundary to OPERABLE status, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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* The CREACS is a shared system with Salem Unit 2

** Alignment only permitted if the Unit with the operable CREACS train is also in Chilled Water System LCO 3.7.10a configuration. Alignment is not permitted if in the LCO 3.7.10c configuration.

LIMITING CONDITION FOR OPERATION (Continued)

d. With one or both series isolation damper(s) on a normal Control Area Air Conditioning System (CAACS) outside air intake or exhaust duct inoperable, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (Refer to ACTION 25 of Table 3.3-6.)

e. With one or both isolation damper(s) on an outside emergency air conditioning air intake duct inoperable, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position and restore the damper(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6 or during movement of irradiated fuel assemblies

a. With one filtration train inoperable, align CREACS for single filtration train operation** within 4 hours, or suspend movement of irradiated fuel assemblies.

b. With CREACS aligned for single filtration train operation with one of the two remaining fans or associated outlet damper inoperable, restore the fan or damper to OPERABLE status within 72 hours, or suspend movement of irradiated fuel assemblies.

c. With two filtration trains inoperable, immediately suspend movement of irradiated fuel assemblies.

d. With the Control Room Envelope boundary inoperable, immediately suspend movement of irradiated fuel assemblies.

e. With one or both series isolation damper(s) on a normal CAACS outside air intake or exhaust duct inoperable, immediately suspend movement of irradiated fuel assemblies until the affected duct is closed by use of at least one isolation damper secured in the closed position. (Refer to ACTION 25 of Table 3.3-6.)

f. With one or both series isolation damper(s) on an outside emergency air conditioning air intake duct inoperable, immediately suspend movement of irradiated fuel assemblies until the affected duct is closed by use of at least one isolation damper secured in the closed position. To resume movement of irradiated fuel assemblies, at least one emergency air intake duct must be operable on each unit.

LIMITING CONDITION FOR OPERATION (Continued)

d. With one or both series isolation damper(s) on a normal Control Area Air Conditioning System (CAACS) outside air intake or exhaust duct inoperable, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (Refer to ACTION 25 of Table 3.3-6.)

e. With one or both isolation damper(s) on an outside emergency air conditioning air intake duct inoperable, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position and restore the damper(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6 or during movement of irradiated fuel assemblies

a. With one filtration train inoperable, align CREACS for single filtration train operation** within 4 hours, or suspend movement of irradiated fuel assemblies.

b. With CREACS aligned for single filtration train operation with one of the two remaining fans or associated outlet damper inoperable, restore the fan or damper to OPERABLE status within 72 hours, or suspend movement of irradiated fuel assemblies.

c. With two filtration trains inoperable, immediately suspend movement of irradiated fuel assemblies.

d. With the Control Room Envelope boundary inoperable, immediately suspend movement of irradiated fuel assemblies.

e. With one or both series isolation damper(s) on a normal CAACS outside air intake or exhaust duct inoperable, immediately suspend movement of irradiated fuel assemblies until the affected duct is closed by use of at least one isolation damper secured in the closed position. (Refer to ACTION 25 of Table 3.3-6.)

f. With one or both series isolation damper(s) on an outside emergency air conditioning air intake duct inoperable, immediately suspend movement of irradiated fuel assemblies until the affected duct is closed by use of at least one isolation damper secured in the closed position. To resume movement of irradiated fuel assemblies, at least one emergency air intake duct must be operable on each unit.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each control room emergency air conditioning system filtration train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating flow through the HEPA filter and charcoal adsorber train(s) and verifying that the train(s) operates with each fan operating for at least 15 minutes.

b. In accordance with the Surveillance Frequency Control Program or prior to return to service (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 system, by:

   1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at a flow rate of 8000 cfm ± 10%.

   2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the ventilation system at a flow rate of 8000 cfm ± 10%.

   3. Verifying within 31 days after removal from the CREACS unit, 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 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the CREACS unit, 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 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

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

   1. Verifying that the pressure drop across the combined HEPA filter and charcoal adsorber bank is ≤ 3.5 inches water gauge while operating the ventilation system at a flow rate of 8000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that on a safety injection test signal or control room intake high radiation test signal, the system automatically actuates in the pressurization mode by opening the outside air supply and diverting air flow through the HEPA filter and charcoal adsorber bank.

3. Deleted.

4. Verifying that on a manual actuation signal, the system will actuate to the required pressurization or recirculation operating mode.

5. Verify each CREACS train has the capability to remove the assumed heat load.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

4.7.6.2 Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Habitability Program (Refer to TS 6.18).

3/4.7.7 AUXILIARY BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 At least two supply fans, and three exhaust fans shall be OPERABLE (*) to maintain the Auxiliary Building at slightly negative pressure.

---------------------------------NOTE--
The intermittent opening of the Auxiliary Building pressure boundary causing a loss of negative pressure may be performed under administrative controls.

APPLICABILITY: At all times

ACTION:

Modes 1 thru 4

a) With one supply fan and/or one exhaust fan inoperable, restore the fan(s) to OPERABLE status within 14 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b) With two supply and/or two exhaust fans inoperable restore at least one inoperable supply and two exhaust fans to operable status within 24 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c) With the Auxiliary Building pressure not maintained slightly negative, restore the Auxiliary Building to slightly negative pressure within the next 4 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During CORE ALTERATIONS

d) With the Auxiliary Building pressure not maintained slightly negative, restore the Auxiliary Building to slightly negative pressure within the next 4 hours or suspend all operations involving CORE ALTERATIONS.

At all times

e) With the Auxiliary Building pressure not maintained slightly negative, suspend all operations involving radioactive gaseous releases via the Auxiliary Building immediately.

(*) One of the supply fans may be considered OPERABLE with its auto start circuit administratively controlled (removed from service) to prevent more than one supply fan from operating at any time.

SURVEILLANCE REQUIREMENTS

4.7.7.1 The above required Auxiliary Building Ventilation System shall be demonstrated OPERABLE by:

a. In accordance with the Surveillance Frequency Control Program by verifying negative pressure in the Auxiliary Building.

b. In accordance with the Surveillance Frequency Control Program by starting each fan, from the control room, and verifying that each fan operates for at least 15 minutes.

c. In accordance with the Surveillance Frequency Control Program by verifying that the System starts following a Safety Injection Test Signal.

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3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.8.1 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 ≥ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.1.1 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.

4.7.8.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequency described below.

a. Sources in use (excluding startup sources and fission detectors previously subjected to core flux) - In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive materials.

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

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 and following repair or maintenance to the source or detector.

4.7.8.1.3 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 $\geq$ 0.005 microcuries of removable contamination.

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours, replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9 on the supported component or declare the supported system inoperable and follow appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Testing Program for examination, testing and service life monitoring of snubbers.

Pages 3/4 7-30 through 3/4 7-32 have been deleted.

3/4.7.10 CHILLED WATER SYSTEM - AUXILIARY BUILDING SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10 The chilled water system loop which services the safety-related loads in the Auxiliary Building shall be OPERABLE with one of the following configurations:

	a	b	c
Configuration	1. Three OPERABLE chillers and, 2. Two OPERABLE chilled water pumps	1. Two OPERABLE chillers and, 2. Two OPERABLE chilled water pumps	1. Three OPERABLE chillers and, 2. Two OPERABLE chilled water pumps from either Unit 1 or Unit 2 (Units Cross-tied)(2)
APPLICABILITY	1. ALL MODES and during movement of irradiated fuel assemblies	1. From November 1 through April 30 in ALL MODES and during movement of irradiated fuel assemblies# 2. The Unit 1 Emergency Control Air Compressor (ECAC) is isolated from the chilled water system 3. Chilled water flow to the third chiller that is not in service is isolated(1) 4. Control Room Emergency Air Conditioning System (CREACS) alignment a. BOTH CREACS trains OPERABLE, no additional chilled water heat load removal required, OR b. Single CREACS train OPERABLE (TS 3.7.6.1 ACTION a.) the following restrictions apply: i. Alignment only permitted to Unit 2 ii. Unit 2 must be in the LCO 3.7.10a configuration iii. Non-essential heat loads are isolated from the chilled water system on BOTH Units	1. From November 1 through April 30 in ALL MODES and during movement of irradiated fuel assemblies## 2. The Unit 1 and Unit 2 ECACs are isolated from the chilled water system 3. Non-Essential heat loads are isolated from the chilled water system on BOTH Units 4. BOTH CREACS trains are operable per TS 3.7.6.1 (single filtration train alignment is not permitted) 5. Unit chilled water cross-tie valves are OPEN 6. Administrative controls are in place for the Unit providing the required components to notify the other Unit if a chiller or pump becomes inoperable

The LCO 3.7.10b configuration may only be used for periods of 60 contiguous days. The 60-contiguous days does not apply for LCO 3.7.10b entry to support the replacement of all 6 original chillers (Units 1 and 2).

The LCO 3.7.10c configuration may only be used for periods of 45 contiguous days.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION[3]: MODES 1, 2, 3, and 4

a. With one of the required chillers inoperable:

   1. Remove[4] the appropriate non-essential heat loads from the chilled water system within 4 hours and;

   2. Restore the chiller to OPERABLE status within 14 days or;

   3. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two of the required chillers inoperable[5][6]:

   1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

   2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 2 train within 4 hours and;

   3. Restore at least one chiller to OPERABLE status within 72 hours or;

   4. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one chilled water pump inoperable, restore the chilled water pump to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION[3]: MODES 5 and 6 or during movement of irradiated fuel assemblies. *

a. With one of the required chillers inoperable:

   1. Remove[4] the appropriate non-essential heat loads from the chilled water system within 4 hours and;

   2. Restore the chiller to OPERABLE status within 14 days or;

   3. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

LIMITING CONDITION FOR OPERATION (Continued)

b. With two of the required chillers inoperable(5)(6):

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 2 train within 4 hours and;

3. Restore at least one chiller to OPERABLE status within 72 hours or;

4. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

c. With one chilled water pump inoperable, restore the chilled water pump to OPERABLE status within 7 days or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.10 The chilled water loop which services the safety-related loads in the Auxiliary Building shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each manual valve in the chilled water system flow path servicing safety related components 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 each automatic valve actuates to its correct position on a Safeguards Initiation signal.

c. In accordance with the Surveillance Frequency Control Program by verifying that each chiller starts and runs.

d. When in the LCO 3.7.10b configuration verify once per 24 hours:

(i) The Unit 1 ECAC is isolated from the chilled water system,
(ii) Chilled water flow is isolated to the third chiller that is not in service and,
(iii) If CREACS is in single filtration alignment verify non-essential heat loads are isolated from the chilled water system on BOTH Units.

e. When in the LCO 3.7.10c configuration verify once per 24 hours:

(i) The Unit 1 and Unit 2 ECACs are isolated from the chilled water system,
(ii) Non-essential heat loads are isolated from the chilled water system and,
(iii) Cross-tie valves are verified OPEN.

SURVEILLANCE REQUIREMENTS (Continued)

NOTES

(1) When transitioning from the LCO 3.7.10b to the LCO 3.7.10a configuration, the chiller may be un-isolated (restored to service) under administrative controls

(2) The LCO 3.7.10c (Cross-Tied) configuration is common to both Units; either Unit 1 chilled water components are required operable, OR Unit 2. A combination of both Units chilled water components is not permitted. When transitioning from the LCO 3.7.10c configuration to either the LCO 3.7.10a or LCO 3.7.10b configurations, chilled water components may be restored to service under administrative controls

(3) When in the LCO 3.7.10c configuration ACTIONs are applicable for both Units

(4) When in the LCO 3.7.10c configuration this ACTION has already been implemented

(5) When in the LCO 3.7.10b configuration, implement Action b.2 AND Action b.4 OR transition to the LCO 3.7.10c configuration

(6) When in LCO 3.7.10c configuration, proceed directly to Action b.4

* During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components are not considered to be inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable. This is not applicable to the LCO 3.7.10c configuration.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.11 The fuel storage pool boron concentration shall be ≥ 800 ppm.

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

ACTION:

With fuel storage pool boron concentration not within limit:

a. Immediately suspend movement of fuel assemblies in the fuel storage pool and

b. Initiate action to:

   1. immediately restore fuel storage pool boron concentration to within limit or

   2. immediately perform a fuel storage pool verification.

c. LCO 3.0.3 is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11 Verify the fuel storage pool boron concentration is within limit in accordance with the Surveillance Frequency Control Program.

3/4.7.12 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.7.12 The combination of initial enrichment, burnup, and Integral Fuel Burnable Absorber (IFBA) of each fuel assembly stored in Region 1 or Region 2, shall be within the acceptable limits described in the surveillance requirements below.

APPLICABILITY: When any fuel assembly is stored in Region 1 or Region 2 of the spent fuel storage pool.

ACTION:

If the requirements of the LCO are not met:

a. Immediately verify the fuel storage boron concentration meets the requirements of TS 3.7.11 and

b. Immediately initiate action to move the non-complying fuel assembly to a location that complies with the surveillance requirements.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 Prior to storing fuel assemblies in Region 1, verify by administrative means that the fuel assemblies meet one of the following storage constraints:

a. Unirradiated fuel assemblies with a maximum enrichment of 4.25 wt% U-235 have unrestricted storage.

b. Unirradiated fuel assemblies with enrichments greater than 4.25 wt% U-235 and less than or equal to 5.0 wt% U-235, that do not contain IFBA pins, may only be stored in the peripheral cells facing the concrete wall.

c. Unirradiated fuel assemblies with enrichments (E) greater than 4.25 wt% U-235 and less than or equal to 5.0 wt% U-235, which contain a minimum number of IFBA pins have unrestricted storage. This minimum number of IFBA pins shall have an equivalent reactivity hold-down which is greater than or equal to the reactivity hold-down associated with N IFBA pins, at a nominal 2.35 mg B-10/linear inch loading (1.5x), determined by the equation below:

$$N = 42.67\ (E - 4.25)$$

d. Irradiated fuel assemblies with enrichments (E) greater than 4.25 wt% U-235 and less than or equal to 5.0 wt%, that have attained the minimum burnup (BU) as determined by the equation below, have unrestricted storage.

$$BU\ (MWD/kg\ U) = -26.212 + 6.1677E$$

4.7.12.2 Prior to storing fuel assemblies in Region 2, verify by administrative means that the fuel assemblies meet one of the following storage constraints:

a. Unirradiated fuel assemblies with a maximum enrichment of 5.0 wt% U-235 may be stored in a checkerboard pattern with intermediate cells containing only water or non-fissile bearing material.

b. Unirradiated fuel assemblies with a maximum enrichment (E) of 5.0 wt% U-235 may be stored in the central cell of any 3x3 array of cells provided the surrounding eight cells are empty or contain fuel assemblies that have attained the minimum burnup (BU) as determined by the equation below.

$$BU\ (MWD/kg\ U) = -15.48 + 17.80E - 0.7038E^2$$

In this configuration, none of the nine cells in any 3x3 array shall be common to cells in any other similar 3x3 array. Along the rack periphery, the concrete wall is equivalent to 3 outer cells in a 3x3 array.

c. Irradiated fuel assemblies with a maximum enrichment (E) of 5.0 wt% U-235 that have attained the minimum burnup (BU) as determined by the equation below, have unrestricted storage.

$$BU\ (MWD/kg\ U) = -32.06 + 25.21E - 3.723E^2 + 0.3535E^3$$

d. Irradiated fuel assemblies with a maximum enrichment (E) of 5.0 wt% U-235 that have attained the minimum burnup (BU) as determined by the equation below, may be stored in a peripheral cell facing the concrete wall.

$$BU\ (MWD/kg\ U) = -25.56 + 15.14E - 0.602E^2$$

3/4.7.13 MAIN FEEDWATER ISOLATION VALVES (FIVS), MAIN FEEDWATER REGULATING VALVES (FRVS), FRV BYPASS VALVES (FRVBVS), AND STEAM GENERATOR FEEDWATER PUMP (SGFP) TURBINE STEAM STOP VALVES

LIMITING CONDITION FOR OPERATION

3.7.13 Four Main FIVs, four Main FRVs, four Main FRVBV, and four SGFP turbine steam stop valves shall be OPERABLE.

APPLICABILITY:

For the FIV in each main feedwater line:

MODES 1, 2, and 3 except when:

a. The FIV is closed and deactivated; or
b. The associated FRV and FRVBV are closed and deactivated; or
c. The associated main feedwater line is isolated by a closed manual valve

For the FRV in each main feedwater line:

MODES 1, 2, and 3 except when:

a. The FRV is closed and deactivated; or
b. The associated FIV is closed and deactivated; or
c. The associated main feedwater line is isolated by a closed manual valve

For the FRVBV in each main feedwater line:

MODES 1, 2, and 3 except when:

a. The FRVBV is closed and deactivated; or
b. The associated FIV is closed and deactivated; or
c. The associated main feedwater line is isolated by a closed manual valve

For each SGFP Turbine Steam Stop Valve:

MODES 1, 2, and 3 except when:

a. The SGFP Turbine Steam Stop Valve is closed and deactivated; or
b. The associated steam supply to the SGFP turbine is isolated by a closed manual valve; or
c. The SGFP feedwater flow path is isolated

3/4.7.13 MAIN FEEDWATER ISOLATION VALVES (FIVS), MAIN FEEDWATER REGULATING VALVES (FRVS), FRV BYPASS VALVES (FRVBVS), AND STEAM GENERATOR FEEDWATER PUMP (SGFP) TURBINE STEAM STOP VALVES

LIMITING CONDITION FOR OPERATION (continued)

ACTION:

---NOTE---
Separate Condition Entry is allowed for each valve
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a. With one or more FIVs inoperable, restore the inoperable FIV(s) to OPERABLE status or close or isolate the inoperable FIV(s) within 72 hours; verify the inoperable FIV(s) is closed or isolated once per 7 days.

b. With one or more FRVs inoperable, restore the inoperable FRV(s) to OPERABLE status or close or isolate the inoperable FRV(s) within 72 hours; verify the inoperable FRV(s) is closed or isolated once per 7 days.

c. With one or more FRVBV(s) inoperable, restore the inoperable FRVBV(s) to OPERABLE status or close or isolate the inoperable FRVBV(s) within 72 hours; verify the inoperable FRVBV(s) is closed or isolated once per 7 days.

d. With one or more SGFP turbine steam stop valves inoperable, restore the inoperable SGFP turbine stop valve(s) to OPERABLE status or isolate the associated steam supply to the SGFP turbine or isolate the SGFP flow path within 72 hours; verify that the inoperable SGFP steam stop valve is isolated or the SGFP flow path is isolated once per 7 days.

e. With two (2) valves in the same feedwater flowpath inoperable resulting in a loss of feedwater isolation capability for a flow path, restore at least one valve to OPERABLE status or isolate the affected flow path within 8 hours.

f. With the required ACTION requirements above not met, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.13.1 Each FIV, FRV, FRVBV and SGFP turbine steam stop valve shall be demonstrated OPERABLE by determining the isolation time of each valve to be within limits when tested pursuant to the INSERVICE TESTING PROGRAM.

4.7.13.2 In accordance with the Surveillance Frequency Control Program, verify each FIV, FRV, FRVBV and SGFP turbine steam stop valve actuates to the isolation position on an actual or simulated actuation signal.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 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 A.C. circuits between the offsite transmission network and the onsite Class 1E distribution system (vital bus system), and

b. Three separate and independent diesel generators with:

    1. Separate day tanks containing a minimum volume of 130 gallons of fuel, and

    2. A common fuel storage system consisting of two storage tanks, each containing a minimum volume of 23,000 gallons of fuel, and two fuel transfer pumps.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With an independent A.C. circuit of the above required A.C. electrical power sources inoperable:

    1. Demonstrate the OPERABILITY of the remaining independent A.C. circuit by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; and

    2. Within 24 hours, declare required systems or components with no offsite power available inoperable when a redundant required system or component is inoperable, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

    3. Restore the inoperable independent A.C. circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of the above required A.C. electrical power sources inoperable:

    1. Demonstrate the OPERABILITY of the independent A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and

    2. Within 4 hours, declare required systems or components supported by the inoperable diesel generator inoperable when a required redundant system or component is inoperable, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

ACTION (Continued)

3. Determine the two remaining OPERABLE diesel generators are not inoperable due to common cause failure or perform Surveillance Requirement 4.8.1.1.2.a.2 within 24 hours. If the diesel generator is inoperable for preventive maintenance, the two remaining OPERABLE diesel generators need not be tested nor the OPERABILITY evaluated; and

4. In any case:

   a) Restore the inoperable diesel generator to OPERABLE status:

      1. Within 72 hours, or

      2. Within 14 days if the Supplemental Power Source (SPS) is available within 72 hours and verified once per 12 hours thereafter. If at any time the availability of the SPS cannot be met, either:

         a. Restore the SPS to available status or restore the diesel generator to OPERABLE status within 72 hours from entry into 3.8.1.1 Action b, or

         b. If 3.8.1.1 Action b has been entered for ≥ 48 hours, restore the SPS to available status or restore the diesel generator to OPERABLE status within 24 hours,

      Otherwise,

      3. Be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the next 30 hours.

c. With one independent A.C. circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining independent A.C. circuit by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; demonstrate the OPERABILITY of the remaining OPERABLE diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.2 within 8 hours; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two independent A.C. circuits and three diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION (Continued)

d. With two of the above required independent A.C. circuits inoperable:

   1. Demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.2 within 8 hours, unless the diesel generators are already operating; and

   2. Within 12 hours, declare required systems or components supported by the inoperable offsite circuits inoperable when a required redundant system or component is inoperable, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

   3. Restore at least one of the inoperable independent A.C. circuits to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours; and

   4. With only one of the independent A.C. circuits OPERABLE, restore the other independent A.C. circuit to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two or more of the above required diesel generators inoperable, demonstrate the OPERABILITY of two independent A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least two of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore three diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With one of the above required fuel transfer pumps inoperable, either restore it to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

g. With one of the above required fuel storage tanks inoperable, either restore it to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

h. LCO 3.0.4.b is not applicable to DGs.

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SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent A.C. circuits between the offsite transmission network and the onsite Class 1E distribution system (vital bus system) shall be:

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, power availability, and

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during shutdown by transferring (manually and automatically) vital bus supply from one 13/4 kv transformer to the other 13/4 kv transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

1. Verifying the fuel level in its day tank.

2. Verifying the diesel generator starts from standby conditions* and achieves ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of 60 ± 1.2 Hz.

   Subsequently, verifying the generator is synchronized with voltage maintained ≥ 3910 and ≤ 4580 volts, gradually loaded to 2340-2600 kw**, and operates at a load of 2340-2600 kw for greater than or equal to 60 minutes.

3. Verifying the diesel generator is aligned to provide standby power to the associated vital bus.

b. 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 one hour by checking for and removing accumulated water from the day tanks.

c. In accordance with the Surveillance Frequency Control Program by verifying the diesel generator starts from standby conditions* and achieves ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of 60 ± 1.2 Hz.

   The generator shall be synchronized to its emergency bus with voltage maintained ≥ 3910 and ≤ 4580 volts, loaded to 2340-2600** kw in less than or equal to 60 seconds, and operate at a load of 2340-2600 kw for at least 60 minutes.

   This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.2, may also serve to concurrently meet those requirements.

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

1. DELETED

2. Verifying that, on rejection of a load greater than or equal to 820 kw, the voltage and frequency are restored to ≥ 3910 and ≤ 4400 volts and 60 ± 1.2 Hz within 4 seconds, and subsequently achieves a steady state frequency of ≥ 58.8 and ≤ 60.5 Hz.

3. Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the vital bus and load shedding from the vital bus.

   b) Verifying the diesel starts on the auto-start signal*, energizes the vital bus with permanently connected loads within 13 seconds, energizes the auto-connected 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 of these loads, the steady state voltage and frequency of the vital bus shall be maintained at ≥ 3910 and ≤ 4400 volts and ≥ 58.8 and ≤ 60.5 Hz during this test.

4. Verifying that on an ESF 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 ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of ≥ 58.8 and ≤ 60.5 Hz.

5. Not Used.

6. Simulating a loss of offsite power in conjunction with an ESF actuation test signal, and:

   a) Verifying de-energization of the vital bus and load shedding from the vital bus.

   b) Verifying the diesel starts on the auto-start signal*, energizes the vital bus with permanently connected loads within 13 seconds, energizes the auto-connected emergency (accident) 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 of these loads, the steady state voltage and frequency of the vital bus shall be maintained at ≥ 3910 and ≤ 4400 volts and ≥ 58.8 and ≤ 60.5 Hz during this test.

c) Verifying that all nonessential automatic diesel generator trips (i.e., other than engine overspeed, lube oil pressure low, 4 KV bus differential and generator differential), are automatically bypassed upon loss of voltage on the vital bus concurrent with a safety injection actuation signal.

7. Deleted

8. Verifying that the auto-connected loads to each diesel generator do not exceed the two hour rating of 2860 kw.

9. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power.

e. 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 58.8 Hz in less than or equal to 13 seconds.

f. In accordance with the Surveillance Frequency Control Program, the following test shall be performed within 5 minutes of diesel shutdown after the diesel has operated for at least two hours at 2340-2600 kw**:

Verifying the diesel generator starts and achieves $\geq$ 3910 volts and $\geq$ 58.8 Hz in $\leq$ 13 seconds, and subsequently achieves steady state voltage of $\geq$ 3910 and $\leq$ 4400 volts and frequency of 60 $\pm$ 1.2 Hz.

g. In accordance with the Surveillance Frequency Control Program verifying the diesel generator operates for at least 24 hours*. During the first 2 hours of this test, the diesel generators shall be loaded to 2760-2860 Kw**. During the remaining 22 hours of this test, the diesel generator shall be loaded to 2500-2600 Kw**. The steady state voltage and frequency shall be maintained at $\geq$ 3910 and $\leq$ 4580 volts and 60 $\pm$ 1.2 Hz during this test.

4.8.1.1.3 The diesel fuel oil storage and transfer system shall be demonstrated OPERABLE:

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

1. Verifying the level in each of the above required fuel storage tanks.

2. Verifying that both fuel transfer pumps can be started and transfer fuel from the fuel storage tanks to the day tanks.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program by verifying that a sample of diesel fuel from each of the above required fuel storage tanks is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

4.8.1.1.4 Reports - NOT USED

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* Surveillance testing may be conducted in accordance with the manufacturer's recommendations regarding engine prelube, warm-up and loading (unless loading times are specified in the individual Surveillance Requirements).

** This band is meant as guidance to preclude routine exceedances of the diesel generator manufacturer's design ratings. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NOT USED

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 (vital bus system), and

b. Two separate and independent diesel generators with:

   1. Separate day tanks containing a minimum volume of 130 gallons of fuel, and

   2. A common fuel storage system containing a minimum volume of 23,000 gallons of fuel, and

   3. A fuel transfer pump.

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

ACTION:

a. With one of the above minimum required A.C. electrical power sources not OPERABLE, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, movement of irradiated fuel, and positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

b. With two of the required diesel generators not OPERABLE, suspend all operations involving CORE ALTERATIONS, movement of irradiated fuel, and all operations involving positive reactivity additions, and immediately initiate action to restore one required DG to OPERABLE status.

SURVEILLANCE REQUIREMENTS

---------------------------------------NOTE---------------------------------
The following surveillances are not required to be performed to maintain operability during Modes 5 and 6. These surveillances are: 4.8.1.1.1.b, 4.8.1.1.2.d.2, 4.8.1.1.2.d.3, 4.8.1.1.2.d.4, 4.8.1.1.2.d.6, 4.8.1.1.2.d.9, 4.8.1.1.2.e, 4.8.1.1.2.f, and 4.8.1.1.2.g.

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2, 4.8.1.1.3 (except for requirement 4.8.1.1.3.a.2) and 4.8.1.1.4.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generators:

4 kvolt	Vital Bus # 1A
4 kvolt	Vital Bus # 1B
4 kvolt	Vital Bus # 1C
460 volt	Vital Bus # 1A and associated control centers
460 volt	Vital Bus # 1B and associated control centers
460 volt	Vital Bus # 1C and associated control centers
230 volt	Vital Bus # 1A and associated control centers
230 volt	Vital Bus # 1B and associated control centers
230 volt	Vital Bus # 1C and associated control centers
115 volt	Vital Instrument Bus # 1A and Inverter *
115 volt	Vital Instrument Bus # 1B and Inverter *
115 volt	Vital Instrument Bus # 1C and Inverter *
115 volt	Vital Instrument Bus # 1D and Inverter *

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With less than the above complement of A.C. busses OPERABLE or energized, restore the inoperable bus to OPERABLE and energized status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one inverter inoperable, energize the associated A.C. Vital Bus within 8 hours; restore the inoperable inverter to OPERABLE and energized status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

(*) An inverter may be disconnected from its DC source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided (1) its vital bus is OPERABLE and energized, and (2) the vital busses associated with the other battery banks are OPERABLE and energized.

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two A.C. electrical bus trains shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator with each train consisting of:

1 - 4 kvolt Vital Bus

1 - 460 volt Vital Bus and associated control centers

1 - 230 volt Vital Bus and associated control centers

1 - 115 volt Instrument Bus energized from its respective inverter connected to its respective D.C. bus train.

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

ACTION:

With less than the above complement of A.C. busses and inverters OPERABLE and energized, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement of irradiated fuel assemblies until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses and inverters shall be determined OPERABLE and energized from A.C. sources other than the diesel generators in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

125-VOLT D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be OPERABLE and energized:

TRAIN 1A consisting of 125-volt D.C. bus No. 1A, 125-volt D.C. battery No. 1A and battery charger 1A1.

TRAIN 1B consisting of 125-volt D.C. bus No. 1B, 125-volt D.C. battery No. 1B and battery charger 1B1.

TRAIN 1C consisting of 125-volt D.C. bus No. 1C, 125-volt D.C. battery No. 1C and battery charger 1C1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 125-volt D.C. bus inoperable or not energized, restore the inoperable bus to OPERABLE and energized status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125-volt D.C. battery charger inoperable, restore the inoperable charger to OPERABLE status within 2 hours or connect the backup charger for no more than 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more 125-volt D.C. batteries with one or more battery cell parameters not within the Category A or B limits of Table 4.8.2.3-1:

   1. Verify within 1 hour, that the electrolyte level and float voltage for the pilot cell meets Table 4.8.2.3-1 Category C limits, and

   2. Verify within 24 hours, that the battery cell parameters of all connected cells meet Table 4.8.2.3-1 Category C limits, and

   3. Restore battery cell parameters to Category A and B limits of Table 4.8.2.3-1 within 31 days, and

   4. If any of the above listed requirements cannot be met, comply with the requirements of action f.

d. With one or more 125-volt D.C. batteries with one or more battery cell parameters not within Table 4.8.2.3-1 Category C values, comply with the requirements of action f.

e. With average electrolyte temperature of representative cells less than 65°F, comply with the requirements of action f.

f. Restore the battery to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and power availability.

4.8.2.3.2 Each 125-volt battery and above required charger shall be demonstrated OPERABLE:

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

   1. The parameters in Table 4.8.2.3-1 meet Category A limits.

   2. The overall battery voltage is greater than or equal to 125 volts on float charge.

b. In accordance with the Surveillance Frequency Control Program and once within 24 hours after a battery discharge < 110 V and once within 24 hours after a battery overcharge > 150 V by verifying that the parameters in Table 4.8.2.3-1 meet the Category B limits.

c. In accordance with the Surveillance Frequency Control Program by verifying that:

   1. There is no visible corrosion at terminals or connectors or the connection resistance is:

      ≤ 150 micro ohms for inter-cell connections,
      ≤ 350 micro ohms for inter-rack connections,
      ≤ 350 micro ohms for inter-tier connections,
      ≤ 70 micro ohms for field cable terminal connections, and
      ≤ 2500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-rack connections (including cable resistance) all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

   2. The average electrolyte temperature of the representative cells is above 65°F.

d. 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. Remove visible terminal corrosion and verify cell-to-cell and terminal connections are coated with anti-corrosion material.

3. The connection resistance is:

   ≤ 150 micro ohms for inter-cell connections,
   ≤ 350 micro ohms for inter-rack connections,
   ≤ 350 micro ohms for inter-tier connections,
   ≤ 70 micro ohms for field cable terminal connections, and
   ≤ 2500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-rack connections (including cable resistance), all inter-tier connections (including cable resistance), and all field terminal connections at the battery.

e. In accordance with the Surveillance Frequency Control Program by verifying that the battery charger will supply at least 170 amperes at 125 volts for at least 4 hours.

f. 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.

g. 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. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.3.2.f if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

h. At least once per 12 months, during shutdown, if the battery shows signs of degradation OR has reached 85% of the service life with a capacity less than 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

i. At least once per 24 months, during shutdown, if the battery has reached 85% of the service life with capacity greater than or equal to 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

SURVEILLANCE REQUIREMENTS (Continued)

Table 4.8.2.3-1
Battery Cell Parameters 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 ≤ 1/4 inch above maximum level indication mark (a)	> Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark (a)	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	≥ 2.07 V
Specific Gravity (b) (c)	≥ 1.195	≥ 1.190 AND Average of all connected cells ≥ 1.200	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥ 1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charge provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 3 amps when on float charge.

(c) Or battery charging current is < 3 amps when on float charge. This is acceptable only during a maximum of 7 days following a battery recharge.

ELECTRICAL POWER SYSTEMS

125-VOLT D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

2 - 125-volt D.C. busses, and

2 - 125-volt batteries, each with at least one full capacity charger, associated with each of the above D.C. busses.

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

ACTION:

With less than the above complement of D.C. equipment and busses OPERABLE, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement of irradiated fuel assemblies until the minimum required 125 Volt D.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. busses shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt batteries and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

28-VOLT D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.5 The following D.C. bus trains shall be energized and OPERABLE:

TRAIN 1A consisting of 28-volt D.C. bus No. 1A, 28-volt D.C. battery No. 1A and battery charger 1A1.

TRAIN 1B consisting of 28-volt D.C. bus No. 1B, 28-volt D.C. battery No. 1B and battery charger 1B1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 28-volt D.C. bus inoperable or not energized, restore the inoperable bus to OPERABLE and energized status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one required 28-volt D.C. battery charger inoperable, restore the inoperable battery charger to OPERABLE status within 2 hours or connect the backup charger for no more than 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more 28-volt D.C. batteries with one or more battery cell parameters not within the Category A or B limits of Table 4.8.2.5-1:

   1. Verify within 1 hour, that the electrolyte level and float voltage for the pilot cell meets Table 4.8.2.5-1 Category C limits, and
   2. Verify within 24 hours, that the battery cell parameters of all connected cells meet Table 4.8.2.5-1 Category C limits, and
   3. Restore battery cell parameters to Category A and B limits of Table 4.8.2.5-1 within 31 days, and
   4. If any of the above listed requirements cannot be met, comply with the requirements of action f.

d. With one or more 28-volt D.C. batteries with one or more battery cell parameters not within Table 4.8.2.5-1 Category C values, comply with the requirements of action f.

e. With average electrolyte temperature of representative cells less than 65°F, comply with the requirements of action f.

f. Restore the battery to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.8.2.5.1 Each D.C. bus train shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and power availability.

4.8.2.5.2 Each 28-volt battery and above required charger shall be demonstrated OPERABLE:

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

   1. The parameters in Table 4.8.2.5-1 meet Category A limits.

   2. The overall battery voltage is greater than or equal to 27 volts on float charge.

b. In accordance with the Surveillance Frequency Control Program and once within 24 hours after a battery discharge < 25.7 V and once within 24 hours after a battery overcharge > 35 V by verifying that the parameters in Table 4.8.2.5-1 meet the Category B limits.

c. In accordance with the Surveillance Frequency Control Program by verifying that:

   1. There is no visible corrosion at terminals or connectors or the connection resistance is:

      ≤ 50 micro ohms for inter-cell connections,
      ≤ 200 micro ohms for inter-tier connections,
      ≤ 70 micro ohms for field cable terminal connections, and
      ≤ 500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

   2. The average electrolyte temperature of the representative cells is ≥ 65°F.

d. 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. Remove visible terminal corrosion and verify cell-to-cell and terminal connections are coated with anti-corrosion material.

SURVEILLANCE REQUIREMENTS (Continued)

3. The connection resistance is:

   ≤ 50 micro ohms for inter-cell connections,
   ≤ 200 micro ohms for inter-tier connections,
   ≤ 70 micro ohms for field cable terminal connections, and
   ≤ 500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

e. In accordance with the Surveillance Frequency Control Program by verifying that the battery charger will supply ≥ 150 amperes at ≥ 28 volts for ≥ 4 hours.

f. 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.

g. 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. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.5.2.f if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

h. At least once per 12 months, during shutdown, if the battery shows signs of degradation OR has reached 85% of the service life with a capacity less than 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

i. At least once per 24 months, during shutdown, if the battery has reached 85% of the service life with capacity greater than or equal to 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

TABLE 4.8.2.5-1

BATTERY CELL PARAMETER 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 ≤ 1/4 inch above maximum level indication mark[a]	>Minimum level indication mark and ≤ 1/4 inch above maximum level indication mark[a]	Above top of plates and not overflowing
Float Voltage	≥2.13 V	≥2.13 V	≥2.07 V
Specific Gravity[b][c]	≥1.195	≥1.190 AND Average of all Connected cells ≥1.200	Not more than 0.020 below the average of all connected cells AND Average of all connected cells ≥1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charge provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) Or battery charging current is < 2 amps when on float charge. This is acceptable only during a maximum of 7 days following a battery recharge.

28-VOLT D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.6 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 28-volt D.C. bus, and

1 - 28-volt battery and at least one full capacity charger associated with the above D.C. bus.

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

ACTION:

With less than the above complement of D.C. equipment and busses OPERABLE, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement or irradiated fuel assemblies until the minimum required 28 Volt D.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.2.6.1 The above required 28-volt D.C. bus shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and voltage on the bus.

4.8.2.6.2 The above required 28-volt batteries and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.5.2.

3/4 8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.1 All containment penetration conductor overcurrent protective devices required to provide thermal protection of penetrations shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping either the primary or backup protective device, or racking out or removing the primary or backup device within 72 hours, declare the affected system or component inoperable, and verify the primary or backup protective device to be tripped, or the primary or backup device racked out or removed at least once per 7 days thereafter; or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 All required containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program:

    1. For 4.16 KV reactor coolant pump circuits, by performance of:

        (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 control circuits function as designed.

SURVEILLANCE REQUIREMENTS (Continued)

2. By verifying the OPERABILITY of the required molded case and lower voltage circuit breakers, by selecting and functionally testing a representative sample of at least 10% of all the circuit breakers of that type. Circuit breakers selected for functional testing shall be selected on a rotating basis. The functional test shall consist of injecting a current input at the specified setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during the 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.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of the Reactor Coolant System, the fuel storage pool, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 6 (Only applicable to the refueling canal, the fuel storage pool and refueling cavity when connected to the Reactor Coolant System)

ACTION:

With the requirements of the above specification not satisfied, immediately

a. Suspend CORE ALTERATIONS and

b. Suspend positive reactivity additions and

c. Initiate action to restore boron concentration to within limit specified in the COLR.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1. Verify the boron concentration is within the limit of the COLR in accordance with the Surveillance Frequency Control Program.

REFUELING OPERATIONS

UNBORATED WATER SOURCE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.9.2.1 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTION:

--NOTE--
Separate entry is allowed for each unborated water source isolation valve.

With one or more valves not secured in closed position:

a. Immediately suspend CORE ALTERATIONS, and initiate actions to secure valve in closed position.

and

b. Within 4 hours perform Surveillance Requirement 4.9.1

SURVEILLANCE REQUIREMENTS

4.9.2.1 Verify each valve that isolates unborated water sources is secured in the closed position in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

3.9.2.2 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity additions.

b. With both of the required monitors inoperable; immediately suspend all operations involving CORE ALTERATIONS, positive reactivity additions, and initiate action to restore one source range monitor to OPERABLE status; and perform Surveillance Requirement 4.9.1 once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS, and

--------------------------------------NOTE-------------------------------------

Neutron detectors are excluded from CHANNEL CALIBRATION

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

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CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch inside door is capable of being closed and held in place by a minimum of four bolts, or an equivalent closure device installed and capable of being closed,

b. A minimum of one door in each airlock is capable of being closed

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by the Containment Purge and Pressure-Vacuum Relief Isolation System.

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed by a manual or automatic containment isolation valve in accordance with the Surveillance Frequency Control Program.

4.9.4.2 Once per refueling prior to the start of movement of irradiated fuel assemblies within the containment building, verify the capability to close, within 1 hour, the equipment hatch inside door or an equivalent closure device. Applicable only when the equipment hatch is open during movement of irradiated fuel in the containment building.

4.9.4.3 Verify, in accordance with the Surveillance Frequency Control Program, each required containment purge isolation valve actuates to the isolation position on a manual actuation signal.

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3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 In accordance with the Surveillance Frequency Control Program one RHR loop shall be verified in operation and circulating coolant at a flow rate of:

a. greater than or equal to 1000 gpm, and

b. sufficient to maintain the RCS temperature at less than or equal to 140°F.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops operable, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per the INSERVICE TESTING PROGRAM.

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* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

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WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The 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 thereafter during movements of fuel assemblies or control rods.

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirements of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the fuel storage pool.

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SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length rod either partially or FULLY WITHDRAWN shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ≤ 85% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The below listed surveillance requirements shall be performed in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Surveillances 4.2.2.2 and 4.2.2.3.

b. Surveillances 4.2.3.1 and 4.2.3.2.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.4, and 3.1.3.5 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at ≤ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

NO FLOW TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set ≤ 25% of RATED THERMAL POWER

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint in accordance with the Surveillance Frequency Control Program during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST prior to initiating startup or PHYSICS TESTS.

3/4.11 RADIOACTIVE EFFLUENTS

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RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outdoor 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 listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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 outdoor temporary tank 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.

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* Tanks included in this Specification are those outdoor temporary tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

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RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume.

APPLICABILITY: At all times. *

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously** monitoring the waste gases in the waste gas holdup system with the oxygen monitor. If hydrogen is not measured, the concentration of hydrogen shall be assumed to exceed 4% by volume.

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* Not applicable to portions of the Waste Gas System removed from service for maintenance provided that, the portions removed for maintenance are isolated, and purged of hydrogen to less than 4% by volume.

** If the oxygen monitoring instrumentation is inoperable, operation of the waste gas holdup system may continue provided grab samples are collected in accordance with the Surveillance Frequency Control Program and analyzed within the following 4 hours.

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SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE LOCATION

Salem Generating Station is located in Salem County, New Jersey along the eastern shore of the Delaware River approximately 8 miles southwest of Salem, New Jersey and 18 miles south of Wilmington, Delaware.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 210 feet.

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 3.5 feet.

e. Minimum thickness of concrete floor mat = 16 feet.

f. Nominal thickness of steel liner = 1/4 to 1/2 inch.

g. Net free volume = 2.62×10^6 cubic feet

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DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 47 psig. Containment air temperatures up to 351.3°F are acceptable providing the containment pressure is in accordance with that described in the UFSAR.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

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 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

5.5 DELETED

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The new fuel storage racks are designed and shall be maintained with:

a. A maximum K_{eff} equivalent of 0.95 with the storage racks flooded with unborated water.

b. A nominal 21.0 inch center-to-center distance between fuel assemblies.

c. Unirradiated fuel assemblies with enrichments less than or equal to 4.25 weight percent (w/o) U-235 with no requirements for Integral Fuel Burnable Absorber (IFBA) pins.

d. Unirradiated fuel assemblies with enrichments (E) greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o U-235 which contain a minimum number of Integral Fuel Burnable Absorber (IFBA) pins. This minimum number of IFBA pins shall have an equivalent reactivity hold-down which is greater than or equal to the reactivity hold down associated with N IFBA pins, at a nominal 2.35 mg B-10/linear inch loading (1.5X), determined by the equation below:

$$N = 42.67 (E - 4.25)$$

5.6.1.2 The spent fuel storage racks are designed and shall be maintained with:

a. A maximum K_{eff} equivalent of 0.95 with the storage racks filled with unborated water.

b. A nominal 10.5 inch center-to-center distance between fuel assemblies stored in Region 1 (flux trap type) racks.

c. A nominal 9.05 inch center-to-center distance between fuel assemblies stored in Region 2 (non-flux trap) racks.

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DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 124'8".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1632 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/hr and 200 cooldown cycles at $\leq$ 100°F/hr (pressurizer cooldown at $\leq$ 200°F/hr).	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 542°F. Cooldown cycle - T_{avg} from $\geq$ 542°F to $\leq$ 200°F.
	80 loss of load cycles.	Without immediate turbine or reactor trip.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite Class 1E distribution system.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	200 large step decreases in load.	50% of RATED THERMAL POWER step load decrease with steam dump.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	1 main reactor coolant pipe break.	Break in a reactor coolant pipe > 13.5 inches equivalent diameter.
	Operating Basis Earthquake Design Basis Earthquake	50 cycles 10 cycles; 0.20g horizontal, 0.136g vertical.
	50 leak tests.	Pressurized to ≥ 2485 psig.
	5 hydrostatic pressure tests	Pressurized to ≥ 3107 psig.
Secondary System	1 steam line break	Break in a steam line > 6 inches equivalent diameter.
	5 hydrostatic pressure tests	Pressurized to ≥ 1356 psig.
	10 turbine roll tests	Turbine roll on pump heat resulting in plant cooldown > 100°F/hr.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall facility 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 Salem 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.

6.2.2 FACILITY STAFF

The facility 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 6.2-1;

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, at least one licensed Senior Reactor Operator shall be in the Control Room area at all times.

c. All CORE ALTERATIONS shall be observed and directly supervised by a licensed Senior Reactor Operator who has no other concurrent responsibilities during this operation.

Figure 6.2-1 CORPORATE HEADQUARTERS AND OFF-SITE ORGANIZATION FOR MANAGEMENT AND TECHNICAL SUPPORT

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FIGURE 6.2 - 2 FACILITY ORGANIZATION

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SALEM UNIT 1

WITH UNIT 2 IN MODES 5 OR 6 OR DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SNSS	1[a]	1[a]
SRO	1[b]	none
STA	1[b]	none
NCO	2	1
EO/UO	3	2[c]
Maintenance Electrician	1	none
Rad. Pro. Technician	1[a]	1[a,e]

WITH UNIT 2 IN MODES 1, 2, 3 OR 4		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SNSS	1[a]	1[a]
SRO	1[b]	none
STA	1[b]	none
NCO	2	1
EO/UO	3[d]	1
Maintenance Electrician	1[a]	none
Rad. Pro. Technician	1[a]	1[a]

a/ Individual may fill the same position on Unit 2.

b/ Individual who fulfills the STA requirement may fill the same position on Unit 2. The STA, if a licensed SRO, may concurrently fill the SRO position on one unit; the other unit also requires a qualified SRO on shift.

c/ One of the two required individuals may fill the position on Unit 2, such that there are a total of three EO/UO's for both units.

d/ One of the three required individuals may fill the same position of Unit 2, such that there are a total of five EO/UO's for both units.

e/ Not needed if both reactors are de-fueled.

TABLE 6.2-1 (Continued)

SNSS - Senior Nuclear Shift Supervisor with a Senior Reactor Operator License on both units.

SRO - Individual with a Senior Reactor Operator License on both units (normally, a Nuclear Shift Supervisor).

NCO - Nuclear Control Operator with a Reactor Operator License on both units.

STA - Shift Technical Advisor (if licensed as SRO, may be assigned duties as a Nuclear Shift Supervisor).

EO/UO - Equipment Operator or Utility Operator.

Except for the Senior Nuclear Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate the unexpected absence of on-duty shift crew members provided that immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewperson's being late or absent.

During any absence of the Senior Nuclear Shift Supervisor from the Control Room area while the unit is in any MODE, an individual with a valid SRO License shall be designated to assume the Control Room command function.

6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall serve in an advisory capacity to the Senior Nuclear Shift Supervisor on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.2.3.2 The Shift Technical Advisor shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions as specified in the PSEG Nuclear Quality Assurance Topical Report.

6.3.2 The Operations Manager or Assistant Operations Manager shall hold an SRO license. The Senior Nuclear Shift Supervisors and Nuclear Shift Supervisors shall each hold a senior reactor operator license. The Nuclear Control Operators shall hold reactor operator licenses.

6.3.3 The Operations Manager shall meet one of the following:

1) Hold an SRO license, or

2) Have held an SRO license for a similar unit (PWR), or

3) Have been certified at an appropriate simulator for equivalent senior operator knowledge.

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6.5 <u>REVIEW AND AUDIT</u> (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 and/or a report submitted pursuant to the requirements of Section 50.73 to 10CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Station Operations Review Committee (SORC) and the resultant Licensee Event Report 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 unit shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The senior corporate nuclear officer and senior management position with responsibility for independent nuclear safety assessment activities and quality program oversight shall be notified within 24 hours.

c. 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 facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. 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 14 days of the violation.

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. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring.

6.8.2 Each procedure and administrative policy of 6.8.1 above, except 6.8.1.d and 6.8.1.e, 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.d and 6.8.1.e 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 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented and receives the same level of review and approval as the original procedure within 14 days of implementation.

6.8.4 The following programs shall be 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 (recirculation spray, safety injection, chemical and volume control, gas stripper, recombiners, ...). The program shall include the following:

(i) Preventative maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements 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 areas under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analyses equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and the control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, including monitoring at the discharge of the condensate pumps for evidence of condenser in-leakage.

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for off-control-point chemistry conditions,

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring

e. Deleted

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 to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after May 7, 2001, shall be performed no later than May 7, 2016.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% 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 L_a. 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 L_a for Type B and Type C tests and less than or equal to 0.75 L_a 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) Seal leakage rate less than or equal to 0.01 La per hour when the gap between the door seals is pressurized to 10.0 psig.

Test frequencies and applicable extensions will be controlled by the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 will be applied 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 the members of the public from radioactive effluents as low as reasonable 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:

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 concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR 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 each 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 92-day period would exceed a suitable fraction 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,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary conforming to 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 each unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50,

10) 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.

6.8.4.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 effluent monitoring program and modeling of 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 the census, and

3) Participation in a 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.

6.8.4.i Steam Generator (SG) Program

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gallon per minute per SG.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate plugging criteria shall be applied as an alternative to the 40% depth based criteria:

1. Tubes with service-induced flaws located greater than 15.21 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.21 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except for any portions of the tube that are exempt from inspection by alternate repair criteria, and that may satisfy the applicable tube plugging criteria.

The portion of the tube below 15.21 inches from the top of the tubesheet is excluded from this requirement.

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 54 effective full power months, which defines the inspection period. If none of the SG tubes have ever experienced cracking other than in regions that are exempt from inspection by alternate repair criteria and the SG inspection was performed with enhanced probes, the inspection period may be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of any type equivalent to or better than array probe technology. The enhanced probes shall be used from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except any portions of the tube that are exempt from inspection by alternate repair criteria. If there are regions where enhanced probes cannot be used, the tube inspection techniques shall be capable of detecting all forms of existing and potential degradation in that region.

3. If crack indications are found in portions of the SG tube excluding any region that is exempt from inspection by alternate repair criteria, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage, but may be deferred to the following refueling outage if 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage.

6.8.4.j Deleted

6.8.4.k Reactor Coolant Pump Flywheel Inspection Program

In addition to the requirements of the ISI Program, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

6.8.4.l 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.

6.8.4.m Snubber Testing Program

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

(i) This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.

(ii) The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval.

(iii) The program shall, as allowed by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements" and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants" in lieu of Section XI of the ASME BPV Code ISI requirements for snubbers, or meet authorized alternatives pursuant to 10 CFR 50.55a(a)(3).

(iv) The 120-month program updates shall be made in accordance with 10 CFR 50.55a (including 10 CFR 50.55a(b)(3)(v)) subject to the limitations and modifications listed therein.

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 U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the USNRC Administrator, Region I, 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 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 plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of 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 power operation), supplementary reports shall be submitted at least every three 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.

* 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.

6.9.1.5 Reports required on an annual basis shall include:

a. DELETED

b. DELETED

c. The results of any specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 DELETED

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 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 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operation shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objective 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 50.

* 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; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.4,

2. Control Bank Insertion Limits for Specification 3/4.1.3.5,

3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,

4. Heat Flux Hot Channel Factor, F_Q, its variation with core height, K(z), and Power Factor Multiplier PF_{xy}, Specification 3/4.2.2, and

5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

6. Refueling boron concentration per Specification 3.9.1

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (W Proprietary), Methodology for Specifications listed in 6.9.1.9.a.

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference.

3. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor.

4. WCAP-10266-P-A, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor.

5. WCAP-12472-P-A, BEACON – Core Monitoring and Operations Support System, (W Proprietary).

6. CENPD-397-P-A, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.9.1.10 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with the Specification 6.8.4.i, “Steam Generator (SG) Program.” The report shall include:

a. The scope of inspections performed on each SG,

b. Then nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;

4. The number of tubes plugged during the inspection outage; and

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results.

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG,

f. The results of any SG secondary side inspections;

g. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

h. The calculated accident induced leakage rate from the portion of the tubes below 15.21 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.16 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined,

i. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

6.9.1.11 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, POPS enable temperature, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.1.2.3, "Charging Pump-Shutdown"
2. Specification 3.4.1.3, “Reactor Coolant System Shutdown”
3. Specification 3.4.1.4, “Reactor Coolant System Cold Shutdown"
4. Specification 3/4.4.9.1, "RCS Pressure/Temperature Limits"
5. Specification 3.4.9.3, "Overpressure Protection Systems"
6. Specification 3/4.5.3, “ECCS Subsystems - Tavg < 350°F”

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC.

1. WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May, 2004
2. WCAP-18124-NP-A, Rev 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2019, may be used as an alternative to Section 2.2 of WCAP-14040-A Rev. 4.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplements thereto.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

6.9.4 When a report is required by ACTION 1, 4, 8 or 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

6.10 RECORD RETENTION

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.

6.10.1 The following records shall be retained for at least five 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. DELETED

f. Records of changes made to Operating Procedures required by Specification 6.8.1.

g. Records of radioactive shipments.

h. Records of sealed source and fission detector leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

j. Records of reviews performed for changes made to procedures or reviews of tests and experiments, pursuant to 10CFR50.59.

6.10.2 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, pursuant to 10CFR50.59.

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 facility components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections.

i. Records of Quality Assurance activities required by the QA Manual.

j. DELETED

k. Records of SORC meetings and activities of the Nuclear Review Board (and activities of its predecessor, the Offsite Safety Review (OSR) staff.)

l. Records for Environmental Qualification which are covered under the provisions of Paragraph 6.16.

m. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

n. Records of secondary water sampling and water quality.

o. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

p. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

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)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Changes to the PCP:

1. Shall be documented and records of review performed shall be retained as required by Specification 6.10.3p. This documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b) 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.

2. Shall become effective after review and acceptance by the SORC and the approval of the Plant Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3p. This documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b) 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.

2. Shall become effective after review and acceptance by the SORC and the approval of the Plant Manager.

3. 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 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee initiated major changes to the radioactive waste system (liquid, gaseous and solid):

1. Shall be reported to the Commission in the UFSAR for the period in which the evaluation was reviewed by (SORC). The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10CFR50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the (SORC).

2. Shall become effective upon review and acceptance by the SORC.

6.16 ENVIRONMENTAL QUALIFICATION

6.16.1 All safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-70 dated October 24, 1980.

6.16.2 Complete and auditable records shall be available and maintained at a central location which describe the environmental qualification method used for all safety related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.17 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 6.17.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.

6.18 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 Air Conditioning System (CREACS), 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.

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 CREACS, operating at the flow rate required by the Surveillance Requirements, at a frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 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 Surveillance Requirements 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. DPR-70

SALEM GENERATING STATION UNIT 1

DOCKET NO. 50-272

AND

FACILITY OPERATING LICENSE NO. DPR-75

SALEM GENERATING STATION UNIT 2

DOCKET NO. 50-311

PSEG NUCLEAR LLC

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

SALEM GENERATING STATION
UNIT NOS. 1 AND 2

ENVIRONMENTAL PROTECTION PLAN
(NONRADIOLOGICAL)

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Amendment No. 100

AUG 2 1 1989

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 NJPDES permit.

2.0 Environmental Protection Issues

In the FES, dated April 1973, the staff considered the environmental impacts associated with the operation of Salem Generating Station Unit Nos. 1 and 2. Certain environmental issues were identified which required study or license conditions to resolve and to assure adequate protection of the environment. The Appendix B Environmental Technical Specifications (ETS) issued with the operating license included discharge restrictions and monitoring programs related to aquatic and terrestrial resources.

2.1 Aquatic Issues

Requirements for study of station intake and discharges effects were removed from the ETS by License Amendments 51 (Unit 1) and 18 Unit 2, dated March 14, 1983 and March 11, 1983, respectively. These issues now are addressed by the effluent limitations and monitoring requirements contained in the effective NJPDES Permit No. NJ0005622 issued by the State of New Jersey, and by the determination of the State of New Jersey on the Section 316(a) and (b) demonstration submitted by licensee. The NRC will rely on the State for regulation of matters involving water quality and aquatic biota.

2.2 Terrestrial Issues

Requirements for study of station effects on terrapins and raptors have been met.

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 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.

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* 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) as 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.

3.2 Reporting Related to the NJPDES Permit or the State Certification

The NRC shall be provided with a copy of the current NJPDES permit or State certification within 30 days of approval. Changes to the NJPDES permit or State certification shall be reported to the NRC within 30 days of the date the change is approved.

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. If an event is reportable under 10 CFR 50.72, then a duplicate immediate report under this Subsection is not required. However, a written report is required in accordance with Section 5.4.2.

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 Nuclear Regulatory Commission (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 threatened or endangered species, decisions made by the National Marine Fisheries Service (NMFS) under the authority of the Endangered Species Act, for any requirements pertaining to aquatic monitoring.

In accordance with Section 7(a) of the Endangered Species Act, the National Marine Fisheries Service has issued multiple Section 7 Consultation Biological Opinions related to the operation of Salem Unit 1 and 2 Generating Stations concluding that "...continued operation is not likely to jeopardize the continued existence of listed species."

PSEG Nuclear LLC shall adhere to the specific requirements within the currently applicable Incidental Take Statement, to the Biological Opinion. Changes to the incidental take statement must be proceeded by consultation between the NRC, as the authorizing agency, and NMFS.

4.2.2 Terrestrial Monitoring

Terrestrial monitoring is not 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 environmental 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 Deleted.

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. DPR-70

PSEG Nuclear LLC and Constellation Energy Generation, LLC shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
192	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.
194	The licensee is authorized to upgrade the initiation circuitry for the power operated relief valves, as described in the licensee's application dated January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented prior to entry into Mode 3 from the current outage for Salem Unit 1.
196	Containment Fan Cooler Units The licensee shall complete all modifications associated with the amendment request concerning Containment Fan Cooler Units (CFCU) response time dated October 25, 1996, as described in the letters supplementing the amendment request dated December 11, 1996, January 28, March 27, April 24, June 3, and June 12, 1997, prior to entry into Mode 3 following refueling outage 12. All modifications made in support of this amendment request and described in the referenced submittals shall be in conformance with the existing design basis for Salem Unit 1, and programmatic controls for tank monitoring instrumentation shall be as described in the letter dated April 24, 1997. Post modification testing and confirmatory analyses shall be as described in the letter dated March 27, 1997. Future changes to the design described in these submittals may be made in accordance with the provisions of 10 CFR 50.59. Further, the administrative controls associated with CFCU operability and containment integrity described in the letters dated March 27, and April 24, 1997 shall not be relaxed or changed without prior staff review until such time as the license has been amended to include the administrative controls as technical specification requirements.	The amendment shall be implemented prior to entry into Mode 3 from the current outage for Salem Unit 1.
198	The licensee shall perform an evaluation of the containment liner anchorage by November 30, 1997, for the loading induced on the containment liner during a Main Steam Line Break event to confirm the assumptions provided in the Preliminary Safety Analysis Report and Updated Final Safety Analysis Report.	The amendment shall be implemented within 30 days from July 17, 1997.