DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE

Renewed License No. NPF-63

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

   A. The application for renewal of the license filed by the Carolina Power & Light Company* (CP&L) for itself complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Shearon Harris Nuclear Power Plant, Unit 1, (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-158 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. 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 analysis that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the 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;

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

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

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*On April 29, 2013, the name "Carolina Power & Light Company" (CP&L) was changed to "Duke Energy Progress, Inc." On August 1, 2015, the name "Duke Energy Progress, Inc." was changed to "Duke Energy Progress, LLC."

F. Duke Energy Progress, LLC* is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Renewed Facility Operating License No. NPF-63, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied;

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

2. Based on the foregoing findings and the Partial Initial Decisions issued by the Atomic Safety and Licensing Board dated February 20, 1985, August 20, 1985, December 11, 1985, and April 28, 1986, regarding this facility and pursuant to approval by the Nuclear Regulatory Commission at a meeting on January 8, 1987, Facility Operating License No. NPF-63, which supersedes the license for fuel loading and low power testing, License No. NPF-53 issued on October 24, 1986, is hereby issued to Duke Energy Progress, LLC. (the licensee) as follows:

A. This license applies to the Shearon Harris Nuclear Power Plant, Unit 1, a pressurized water reactor and associated equipment (the facility) owned and operated by Duke Energy Progress, LLC. The facility is located on the licensee's site in Wake and Chatham Counties, North Carolina, approximately 16 miles southwest of the nearest boundary of Raleigh, and is described in its Final Safety Analysis Report, as supplemented and amended, and in its Environmental Report, as supplemented and amended;

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*Duke Energy Progress, LLC. has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

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

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, Duke Energy Progress, LLC to possess, use, and operate the facility at the designated location in Wake and Chatham Counties, North Carolina, in accordance with the procedures and limitations set forth in this license;

(2) Deleted.

(3) Pursuant to the Act and 10 CFR Part 70, Duke Energy Progress, LLC. 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) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Duke Energy Progress, LLC to receive, possess, and use at any time any byproduct, source and special nuclear material such 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) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Duke Energy Progress, LLC 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;

(6) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Duke Energy Progress, LLC to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein;

(7) Pursuant to the Act and 10 CFR Parts 30 and 40, Duke Energy Progress, LLC to receive, possess and process for release or transfer to the Shearon Harris site such byproduct material as may be produced by the Shearon Harris Energy and Environmental Center;

(8) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Duke Energy Progress, LLC to receive and possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Brunswick Steam Electric Plant, Units 1 and 2, and H. B. Robinson Steam Electric Plant, Unit 2.

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

(1) Maximum Power Level

Duke Energy Progress, LLC, is authorized to operate the facility at reactor Core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 200, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, LLC. shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

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

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company* shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at 15.6.3 Subparts II (1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company* will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

[1]The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.
* On April 29, 2013, the name of “Carolina Power & Light Company” (CP&L) was changed to “Duke Energy Progress, Inc.” On August 1, 2015, the name “Duke Energy Progress, Inc.” was changed to “Duke Energy Progress, LLC.”

(6) Detailed Control Room Design Review (Item I.D.1, Section 18)

Carolina Power & Light* shall submit the final results of the control room surveys prior to startup following the first refueling outage.

(7) Safety Parameter Display System (Section 18.2.1)

Carolina Power & Light Company* shall submit to the NRC for review prior to startup following the first refueling:

(a) The final Validation Test Report,

(b) The resolution of additional human engineering deficiencies identified on the safety parameter display system.

(8) Deleted

(9) Formal Federal Emergency Management Agency Finding

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

(10) Fresh Fuel Storage

The following criteria apply to the storage and handling of new fuel assemblies in the Fuel Handling Building:

(a) The minimum edge-to-edge distance between a new fuel assembly outside its shipping container or storage rack and all other new fuel assemblies shall be at least 12 inches.

(c) New fuel assemblies shall be stored in such a manner that water would drain freely from the assemblies in the event of flooding and subsequent draining of the fuel storage area

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(11) Mitigation Strategy License Condition

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

(12) Control Room Habitability

Upon implementation of Amendment No. 128 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by Surveillance Requirement (SR) 4.7.6.g, in accordance with TS 6.8.4.o.3(i), the assessment of CRE habitability as required by TS 6.8.4.o.3(ii) and the measurement of CRE pressure as required by TS 6.8.4.o.4, shall be considered met. Following implementation:

a) The first performance of SR 4.7.6.g, in accordance with Specification 6.8.4.o.3(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from March 5, 2004, the date of the most recent successful tracer gas test, 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 6.8.4.o.3(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from March 5, 2004, the date of the most recent successful tracer gas test, 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 6.8.4.o.4, shall be within 18 months plus 138 days allowed by SR 4.0.2 as measured from October 13, 2006, the date of the most recent successful pressure measurement test.

D. Exemptions

The facility requires an exemption from Appendix E, Section IV.F.1, which requires that a full participation exercise be conducted within one year before the issuance of a license for full power operation. This exemption is authorized by law and will not endanger life or property or the common defense and security, and certain special circumstances are present. This exemption is, therefore, hereby granted pursuant to 10 CFR 50.12 as follows:

> Shearon Harris Nuclear Power Plant, Unit 1, is exempt from the requirement of 10 CFR Part 50, Appendix E, Section IV.F.1 for the conduct of an offsite full participation exercise within one year before the issuance of the first operating license for full power and prior to operation above 5 percent of rated power, provided that a full participation exercise is conducted before or during March 1987.

The facility is granted an exemption from Paragraph III.D.2(b)(ii) of Appendix J to 10 CFR Part 50 (see SER Section 6.2.6). This exemption is authorized by law and will not endanger life or property or the common defense and security, and certain special circumstances are present. In addition, the facility was previously granted an exemption from the criticality alarm requirements of paragraph 70.24 of 10 CFR Part 70 insofar as this section applies to this license. (See License Number SNM-1939 dated October 28, 1985, which granted this exemption).

E. Physical and Cyber Security (Section 13.6.2.10)

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 the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Guard Training and Qualification Plan" submitted by letter dated October 19, 2004, "Physical Security Plan" and "Safeguards Contingency Plan" submitted by letter dated October 19, 2004 as supplemented by letter dated May 16, 2006.

The licensee 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 licensee's CSP was approved by License Amendment No. 136, as supplemented by changes approved by License Amendment Nos. 140 and 144.

F. Fire Protection Program

Duke Energy Progress, LLC shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the revised license amendment request dated October 9, 2009, supplemented by letters dated February 4, 2010, and April 5, 2010, and approved in the associated safety evaluation dated June 28, 2010. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c) and NFPA 805, and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(1) Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the proposed change may include methods that have been used in the peer-reviewed Fire PRA model, methods that have been approved by the NRC via a plant-specific license amendment or through NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

(a) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10E-7 per year (/yr) for CDF and less than 1×10E-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(2) <u>Other Criteria for Changes that May Be Made to the NFPA 805 Fire Protection Program Without Prior NRC Approval</u>

(a) Changes to NFPA 805 Chapter 3, Fundamental Fire Protection Program Elements and Design Requirements

Prior NRC review and approval is not required for changes to the NFPA 805 Chapter 3 fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard.

The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805 Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805 Chapter 3 elements are acceptable because the alternative is “adequate for the hazard.” Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805 Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

The four specific sections of NFPA 805 Chapter 3 are as follows:

- Fire Alarm and Detection Systems (Section 3.8);
- Automatic and Manual Water-Based Fire Suppression Systems (Section 3.9);
- Gaseous Fire Suppression Systems (Section 3.10); and
- Passive Fire Protection Features (Section 3.11).

This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

(b) Fire Protection Program Changes that Have No More than Minimal Risk Impact

Prior NRC review and approval is not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process, as approved in the NRC safety evaluation dated June 28, 2010, to determine that certain fire protection program changes meet the minimal risk criterion. The licensee shall in all cases ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Unless License Condition F.(2)(b) is met, risk-informed changes to the licensee's fire protection program which involve fire areas that credit incipient detection may not be made without prior NRC review and approval until the Harris Fire PRA model has been modified to incorporate an NRC-accepted method for modeling incipient detection.

(3) <u>Transition License Conditions</u>

(a) Before achieving full compliance with 10 CFR 50.48(c), as specified by Transition License Condition F.(3)(b), risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in License Condition F.(2)(b) above.

(b) The licensee shall implement the following modifications to its facility in order to complete the transition to full compliance with 10 CFR 50.48(c) by December 31, 2010 (note that each modification is listed by Engineering Change (EC) Number, as described in Attachment S of the Shearon Harris NFPA 805 License Amendment Request Transition Report, and outlined in Table 2.8.1-2 of the associated NRC safety evaluation):

- EC 62343
- EC 62820
- EC 68645
- EC 68646
- EC 68648
- EC 68658
- EC 68769
- EC 69501
- EC 69764
- EC 69765
- EC 70027
- EC 70350
- EC 70895
- EC 71147

(c) The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

G. Deleted.

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

I. The Updated Safety Analysis Report supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the Updated Safety Analysis Report required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, CP&L* may make changes to the programs and activities described in the supplement without prior Commission approval, provided that CP&L* evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

J. The Updated Safety Analysis Report supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. Duke Energy Progress, LLC shall complete these activities no later than October 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

K. 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 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 inspection. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

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L. This license is effective as of the date of issuance and shall expire at midnight on October 24, 2046.

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Appendix A – Technical Specifications
2. Appendix B – Environmental Protection Plan
3. Appendix C – Antitrust Conditions
4. Appendix D – Additional Conditions

Date of Issuance: December 17, 2008

NUREG-1240

# Technical Specifications

## Shearon Harris Nuclear Power Plant Unit No. 1

Docket No. 50-400

Appendix "A" to
License No. NPF-63

**Issued by the
U.S. Nuclear Regulatory
Commission**

**Office of Nuclear Reactor Regulation**

January 1987

# INDEX

SECTION 1.0

DEFINITIONS

## 1.0 DEFINITIONS

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

ACTION

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

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 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

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 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.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specification 3.6.3.

b. All equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity 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 position.

CORE OPERATING LIMITS REPORT

1.9.a The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Plant operation within these core operating limits is addressed within the individual specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1.10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation to verify OPERABILITY of alarm and/or trip functions.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979 (or equivalently, Federal Guidance Report No. 11 "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA 520/1-88-020, September 1988).

$\overline{E}$ – AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF 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 the 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.

EXCLUSION AREA BOUNDARY

1.14 The EXCLUSION AREA BOUNDARY shall be that line beyond which the land is not controlled by the licensee to limit access.

FREQUENCY NOTATION

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

1.16 (DELETED)

IDENTIFIED LEAKAGE

1.17 IDENTIFIED LEAKAGE shall be:

a. Leakage, such as that from pump seals or valve packing (except CONTROLLED LEAKAGE), that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of Leakage Detection Systems; or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary to secondary leakage).

INSERVICE TESTING PROGRAM

1.17a The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

MASTER RELAY TEST

1.18 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL

1.20 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 6.9.1.3 and 6.9.1.4.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.22 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 1.2.

PHYSICS TESTS

1.23 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation:

(1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.24 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary to secondary leakage) through a fault in a Reactor Coolant System component body, pipe wall, or vessel wall. Leakage past seals, packing, and gaskets is not PRESSURE BOUNDARY LEAKAGE.

PROCESS CONTROL PROGRAM

1.25 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 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any 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

1.27 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

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2948 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its 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 the 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

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

SHUTDOWN MARGIN

1.31 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster 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. However, with all rod cluster assemblies verified as fully inserted by two independent means, it is not necessary to account for a stuck rod cluster assembly in the SHUTDOWN MARGIN calculation. With any rod cluster assembly not capable of being fully inserted, the reactivity worth of the rod cluster assembly must be accounted for in the determination of SHUTDOWN MARGIN, and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

SITE BOUNDARY

1.32 For these Specifications, the SITE BOUNDARY shall be identical to the EXCLUSION AREA BOUNDARY defined above.

SLAVE RELAY TEST

1.33 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.34 Deleted from Technical Specifications and relocated to the PCP.

SOURCE CHECK

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

STAGGERED TEST BASIS

1.36 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, and

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.38 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.39 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

1.42 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

FREQUENCY NOTATION

| NOTATION | FREQUENCY |
|---|---|
| S | At least once per 12 hours. |
| D | At least once per 24 hours. |
| W | At least once per 7 days. |
| M | At least once per 31 days. |
| Q | At least once per 92 days. |
| SA | At least once per 184 days. |
| R | At least once per 18 months. |
| S/U | Prior to each reactor startup. |
| N.A. | Not applicable. |
| P | Completed prior to each release. |
| SFCP | At the frequency specified in the Surveillance Frequency Control Program |

## TABLE 1.2

## OPERATIONAL MODES

| MODE | REACTIVITY CONDITION, $K_{eff}$ | % RATED 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.

# 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 specified in the COLR; and the following SLs shall not be exceeded:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP DNB correlation for HTP fuel and ≥ 1.12 for the ORFEO-GAIA DNB correlation for GAIA fuel.

b. The peak centerline temperature shall be maintained < [4901 – (1.37 x $10^{-3}$ x (Burnup, MWD/MTU))] °F.

APPLICABILITY: MODES 1 and 2.

ACTION:

If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig except during hydrostatic testing.

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.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock 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.

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

WITH MEASURED RCS FLOW > [290,000 GPM X (1.0 + $C_1$)]

This figure is deleted from Technical Specifications and relocated to the COLR.

APPLICABILITY (Continued)

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

c. With a Reactor Trip System Instrumentation Channel or Interlock inoperable, take the appropriate ACTION shown in Table 3.3-1.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|
| 1. | Manual Reactor Trip | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2. | Power Range, Neutron Flux | | | | | |
| | a. High Setpoint | 4.58 | 3.25 | 0 | ≤ 108% of RTP** See NOTES 7, 8 | ≤ 109.6% of RTP** |
| | b. Low Setpoint | 7.83 | 4.56 | 0 | ≤ 25% of RTP** See NOTES 7, 8 | ≤ 26.8% of RTP** |
| 3. | Power Range, Neutron Flux, High Positive Rate | 2.33 | 0.83 | 0 | ≤ 5% of RTP** with a time constant ≥ 2 seconds See NOTES 7, 8 | ≤ 6.3% of RTP** with a time constant ≥ 2 seconds |
| 4. | Not Used | N/A | N/A | N/A | N/A | N/A |
| 5. | Intermediate Range, Neutron Flux | 17.0 | 8.41 | 0 | ≤ 25% of RTP** | ≤ 30.9% of RTP** |
| 6. | Source Range, Neutron Flux | 17.0 | 10.01 | 0 | ≤ $10^5$ cps | ≤ 1.4 x $10^5$ cps |
| 7. | Overtemperature ΔT | 9.0 | 7.38 | Note 5 | See Note 1 | See Note 2 |
| 8. | Overpower ΔT | 3.33 | 2.43 | 1.3 | See Note 3 | See Note 4 |
| 9. | Pressurizer Pressure-Low | 4.625 | 1.52 | 1.5 | ≥ 1960 psig | ≥ 1948 psig |
| 10. | Pressurizer Pressure-High | 4.625 | 1.52 | 1.5 | ≤ 2385 psig | ≤ 2397 psig |
| 11. | Pressurizer Water Level-High | 8.0 | 3.42 | 1.75 | ≤ 87% of instrument span See NOTES 7, 8 | ≤ 88.5% of instrument span |

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|
| 12. | Reactor Coolant Flow-Low | 3.08 | 1.58 | 0.49 | ≥ 91.7% of loop full indicated flow See NOTES 7, 8 | ≥ 90.6% of loop full indicated flow |
| 13. | Steam Generator Water Level Low-Low | 25.0 | 17.45 | 2.0 | ≥ 25.0% of narrow range instrument span | ≥ 23.5% of narrow range instrument span |
| 14. | Steam Generator Water Level - Low | 8.9 | 5.95 | 2.0 | ≥ 25.0% of narrow range instrument span | ≥ 24.05% of narrow range instrument spa |
| | Coincident With Steam/Feedwater Flow Mismatch | 20.0 | 3.01 | Note 6 | ≤ 40% of full steam flow at RTP** | ≤ 43.1% of full steam flow at RTP** |
| 15. | Undervoltage - Reactor Coolant Pumps | 14.0 | 1.3 | 0.0 | ≥ 5148 volts | ≥ 4920 volts |
| 16. | Underfrequency - Reactor Coolant Pumps | 5.0 | 3.0 | 0.0 | ≥ 57.5 Hz | ≥ 57.3 Hz |
| 17. | Turbine Trip | | | | | |
| | a. Low Fluid Oil Pressure | N.A. | N.A. | N.A. | ≥ 1000 psig | ≥ 950 psig |
| | b. Turbine Throttle Valve Closure | N.A. | N.A. | N.A. | ≥ 1% open | ≥ 1% open |
| 18. | Safety Injection Input from ESF | N.A. | N.A. | N.A. | N.A. | N.A. |

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|
| 19. | Reactor Trip System Interlocks | | | | | |
| | a. Intermediate Range Neutron Flux, P-6 | N.A. | N.A. | N.A. | ≥ 1 x 10-10 amp | ≥ 6 x 10-11 amp |
| | b. Low Power Reactor Trips Block, P-7 | | | | | |
| | 1) P-10 input | N.A. | N.A. | N.A. | ≤ 10% of RTP** | ≤ 12.1% of RTP** |
| | 2) P-13 input | N.A. | N.A. | N.A. | ≤ 10% RTP** Turbine Inlet Pressure Equivalent | ≤ 12.1% RTP** Turbine Inlet Pressure Equivalent |
| | c. Power Range Neutron Flux, P-8 | N.A. | N.A. | N.A. | ≤ 49% of RTP** | ≤ 51.1% of RTP** |
| | d. Power Range Neutron Flux, P-10 | N.A. | N.A. | N.A. | ≥ 10% of RTP** | ≥ 7.9% of RTP** |
| | e. Turbine Inlet Pressure, P-13 | N.A. | N.A. | N.A. | ≤ 10% RTP** Turbine Inlet Pressure Equivalent | ≤ 12.1% RTP** Turbine Inlet Pressure Equivalent |
| 20. | Reactor Trip Breakers | N.A. | N.A. | N.A. | N.A. | N.A. |
| 21. | Automatic Trip and Interlock Logic | N.A. | N.A. | N.A. | N.A. | N.A. |
| 22. | Reactor Trip Bypass Breakers | N.A. | N.A. | N.A. | N.A. | N.A. |

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

The values denoted with [*] are specified in the COLR.

NOTE 1: OVERTEMPERATURE $\Delta T$

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \left(\frac{1}{1+\tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where: $\Delta T$ = Measured $\Delta T$ by RTD Instrumentation;

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = Lead-lag compensator on measured $\Delta T$;

$\tau_1, \tau_2$ = Time constants utilized in lead-lag compensator for $\Delta T$, $\tau_1$ = [*] s, $\tau_2$ = [*] s;

$\frac{1}{1+\tau_3 S}$ = Lag compensator on measured $\Delta T$;

$\tau_3$ = Time constants utilized in the lag compensator for $\Delta T$, $\tau_3$ = [*] s;

$\Delta T_o$ = Indicated $\Delta T$ at RATED THERMAL POWER;

$K_1$ = [*];

$K_2$ = [*]/°F;

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = The function generated by the lead-lag compensator for $T_{avg}$ dynamic compensation;

$\tau_4, \tau_5$ = Time constants utilized in the lead-lag compensator for $T_{avg}$, $\tau_4$ = [*] s, $\tau_5$ = [*] s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

The values denoted with [*] are specified in the COLR.

NOTE 1: (Continued)

| | | |
|---|---|---|
| T | = | Average temperature, °F; |
| $\frac{1}{1+\tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6$ = [*] s; |
| T' | = | Reference $T_{avg}$ at RATED THERMAL POWER (≤ [*]°F); |
| $K_3$ | = | [*]/psig; |
| P | = | Pressurizer pressure, psig; |
| P' | = | [*] psig (Nominal RCS operating pressure); |
| S | = | Laplace transform operator, $s^{-1}$; |

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

(1) For $q_t$ - $q_b$ between [*]% and [*]%, $f_1$ (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;

(2) For each percent that the magnitude of $q_t$ - $q_b$ exceeds [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t$ - $q_b$ exceeds [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 1.35% of $T_{avg}$ span for $T_{avg}$ input; 0.6% of pressurizer pressure span for pressurizer pressure input; and 0.6% of ΔI span for ΔI input.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

The values denoted with [*] are specified in the COLR.

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{(1)}{(1+\tau_3 S)} \leq \Delta T_o \left\{ K_4 - K_5 \frac{(\tau_7 S)}{(1+\tau_7 S)} \frac{(1)}{(1+\tau_6 S)} T - K_6 \left[ T \left( \frac{(1)}{(1+\tau_6 S)} \right) - T'' \right] - f_2(\Delta I) \right\}$$

| | | | |
|---|---|---|---|
| Where: | $\Delta T$ | = | As defined in Note 1, |
| | $\frac{1+\tau_1 S}{1+\tau_2 S}$ | = | As defined in Note 1, |
| | $\tau_1$, $\tau_2$ | = | As defined in Note 1, |
| | $\frac{1}{1+\tau_3 S}$ | = | As defined in Note 1, |
| | $\tau_3$ | = | As defined in Note 1, |
| | $\Delta T_o$ | = | As defined in Note 1, |
| | $K_4$ | = | [*], |
| | $K_5$ | = | [*]/°F for increasing average temperature and [*] for decreasing average temperature, |
| | $\frac{\tau_7 S}{1+\tau_7 S}$ | = | The function generated by the rate-lag compensator for $T_{avg}$ dynamic compensation, |
| | $\tau_7$ | = | Time constants utilized in the rate-lag compensator for $T_{avg}$, $\tau_7$ = [*] s, |
| | $\frac{1}{1+\tau_6 S}$ | = | As defined in Note 1, |
| | $\tau_6$ | = | As defined in Note 1, |

<u>TABLE 2.2-1 (Continued)</u>

<u>TABLE NOTATIONS</u>

The values denoted with [*] are specified in the COLR.

NOTE 3: (Continued)

| | | |
|---|---|---|
| $K_6$ | = | [*]/°F for T > T" and $K_6$ = [*] for T ≤ T", |
| T | = | As defined in Note 1, |
| T" | = | Reference $T_{avg}$ at RATED THERMAL POWER (≤ [*]°F), |
| S | = | As defined in Note 1, and |
| $f_2(\Delta I)$ | = | [*]. |

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input, 1.35% of $T_{avg}$ span for $T_{avg}$ input; and 0.6% of ΔI span for ΔI input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 0.8% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 2.2-1 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in the FSAR. The as-found and as-left tolerances are specified in the Technical Requirements Manual.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

## 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 unless otherwise noted in the ACTION statement.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 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:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours.

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

This specification is not applicable in MODE 5 or 6.

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;

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 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 3.0.1 above for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0.6 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem

## LIMITING CONDITION FOR OPERATION (Continued)

supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

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

## SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation, unless otherwise stated in an individual Surveillance Requirement. Failure to meet a Surveillance Requirement, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified surveillance interval shall be failure to meet the LCO except as provided in SR 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% of the specified surveillance interval.

4.0.3 If it is discovered that a surveillance was not performed within its specified surveillance interval, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, 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 LCO must immediately be declared not met, and the applicable ACTION requirements must be met.

When the surveillance is performed within the delay period and the surveillance criteria are not met, the LCO must immediately be declared not met, and the applicable ACTION requirements must be met.

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.

4.0.5 Deleted

Page 3/4 0-3 has been deleted by Amendment No.

## 3/4.1 REACTIVITY CONTROL SYSTEMS

## 3/4.1.1 BORATION CONTROL

### SHUTDOWN MARGIN – MODES 1 AND 2

### LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the COLR for 3-loop operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the SHUTDOWN MARGIN less than the limit specified in the COLR, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 greater than or equal to the limit specified in the COLR:

a. Within 1 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 untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the untrippable control rod(s);

b. When in MODE 1 or MODE 2 with $K_{eff}$ greater than or equal to 1 at the frequency specified in the Surveillance Frequency Control Program by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. Within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6; and

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors below, with the control banks at the maximum insertion limit of Specification 3.1.3.6:

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (CONTINUED)

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 ±1000 pcm at the frequency specified in the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1d., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If later experience shows adjustment is desirable at approximately 60 EFPD, the adjustment is permissible.

## SHUTDOWN MARGIN MODES – 3, 4, AND 5

### LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 3, 4, AND 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 greater than or equal to the required value:

a. Within 1 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 untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the untrippable control rod(s); and

b. At the frequency specified in 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.

FIGURE 3.1-1

SHUTDOWN MARGIN VERSUS RCS BORON CONCENTRATION

MODES 3, 4, AND 5/DRAINED

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT.

## MODERATOR TEMPERATURE COEFFICIENT

### LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum positive limit shall be less than or equal to +5 pcm/°F for power levels up to 70% RATED THERMAL POWER and a linear ramp from that point to 0 pcm/°F at 100% RATED THERMAL POWER.

APPLICABILITY: Positive MTC Limit – MODES 1 and 2* only**.

Negative MTC Limit – MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the Positive MTC Limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within the Positive MTC 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 of Specification 3.1.3.6; and

   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.

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

*With $k_{eff}$ greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 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 Positive MTC Limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR 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 Negative MTC Limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

## MINIMUM TEMPERATURE FOR CRITICALITY

### LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature ($T_{avg}$) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature ($T_{avg}$) less than 551°F, restore $T_{avg}$ to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

### SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature ($T_{avg}$) shall be determined to be greater than or equal to 551°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 561°F with the $T_{avg}$-$T_{ref}$ Deviation Alarm not reset.

*With $K_{eff}$ greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid tank via either a boric acid transfer pump or a gravity feed connection and a charging/safety injection pump to the Reactor Coolant System if the boric acid tank in Specification 3.1.2.5a. or 3.1.2.6a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging/safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. or 3.1.2.6b. is OPERABLE.

APPLICABILITY: MODES 4*, 5*, and 6*.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header is greater than or equal to 65°F when a flow path from the boric acid tank is used, and

b. At the frequencies specified in 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.

*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F and the reactor vessel head is in place.

## LIMITING CONDITION FOR OPERATION

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

a. The flow path from the boric acid tank via a boric acid transfer pump and a charging/safety injection pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging/ safety injection pumps to the RCS.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) 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 HOT SHUTDOWN within the next 6 hours.

## SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header tank is greater than or equal to 65°F when a flow path from the boric acid tank is used;

b. At the frequencies specified in 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;

c. At the frequencies specified in the Surveillance Frequency Control Program by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal; and

d. At the frequencies specified in the Surveillance Frequency Control Program by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

## CHARGING PUMP - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging/safety injection pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 4*, 5*#, and 6*#.

ACTION:

With no charging/safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

### SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging/safety injection pump shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the reactor coolant system and reactor coolant pump seals, that a differential pressure across the pump of greater than or equal to 2446 psid is developed when tested pursuant to the INSERVICE TESTING PROGRAM.

4.1.2.3.2 All charging/safety injection pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable** by verifying that each pump's motor circuit breaker is secured in the open position prior to the temperature of one or more of the RCS cold legs decreasing below 325°F and at least once per 31 days thereafter, except when the reactor vessel head is removed.

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* A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F and the reactor vessel head is in place.

** An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator or by a manual isolation valve secured in the closed position.

# For periods of no more than 1 hour, when swapping pumps, it is permitted that there be no OPERABLE charging/safety injection pump. No CORE ALTERATIONS or positive reactivity changes are permitted during this time.

## CHARGING PUMPS - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging/safety injection pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging/safety injection pump OPERABLE, restore at least two charging/safety injection pumps to OPERABLE status within 72 hours* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) at 200°F within the next 6 hours; restore at least two charging/safety injection pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

---------------------------------------------------------- NOTE ----------------------------------------------------------

*One charging/safety injection pump train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

### SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging/safety injection pumps shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the Reactor Coolant System and reactor coolant pump seals, that a differential pressure across each pump of greater than or equal to 2446 psid is developed when tested pursuant to the INSERVICE TESTING PROGRAM.

## BORATED WATER SOURCE - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

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

a. The boric acid tank with:

1. A minimum contained borated water volume of 7150 gallons which is ensured by maintaining indicated level of greater than or equal to 23%,
2. A boron concentration within the limits specified in the COLR, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1. A minimum contained borated water volume of 106,000 gallons, which is equivalent to 12% indicated level,
2. A boron concentration within the limits specified in the COLR, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

### SURVEILLANCE REQUIREMENTS

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

a. At the frequency specified in the Surveillance Frequency Control Program by:

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

b. At the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

## 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 Specification 3.1.2.2:

a. The boric acid tank with:

1. A minimum contained borated water volume of 24,150 gallons, which is ensured by maintaining indicated level of greater than or equal to 74%,
2. A boron concentration within the limits specified in the COLR, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1. A minimum contained borated water volume of 436,000 gallons, which is equivalent to 92% indicated level.
2. A boron concentration within the limits specified in the COLR,
3. A minimum solution temperature of 40°F, and
4. A maximum solution temperature of 125°F.

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

ACTION:

a. With the boric acid tank inoperable and being used as one of the above required borated water sources, restore the boric acid tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN specified in the CORE OPERATING LIMITS REPORT (COLR) at 200°F; restore the boric acid tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, 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

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

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

b. At the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is either less than 40°F or greater than 125°F.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more 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 rod misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. Deleted.

d. With one rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 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:

      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

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

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 power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each rod shall be determined to be within the group demand limit by verifying the individual rod positions at the frequency specified in the Surveillance Frequency Control Program 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 rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at the frequency specified in the Surveillance Frequency Control Program.

## TABLE 3.1-1

## ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small 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 Coolant System Pipe Rupture

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

## POSITION INDICATION SYSTEMS - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the shutdown and control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

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

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

   1. Verify that all digital 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 12 steps of each other at least once per 8 hours, or

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

### SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at the frequency specified in 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 Digital Rod Position Indication System at least once per 4 hours.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **.

ACTION:

ACTION:

a. With one of the above required position indicator(s) inoperable, either restore the indicator to OPERABLE within 8 hours or open the Reactor Trip System breakers.

b. With more than one of the above required position indicators inoperable, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at the frequency specified in the Surveillance Frequency Control Program.

*With the Reactor Trip System breakers in the closed position.

**See Special Test Exceptions Specification 3.10.5.

## ROD DROP TIME

### LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual shutdown and control rod drop time from the fully withdrawn position 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 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any 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.

b. With the rod drop times within limits but determined with two reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

### SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of shutdown and control 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. At the frequency specified in the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn as specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn as specified in the COLR:

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

b. At the frequency specified in the Surveillance Frequency Control Program thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With $K_{eff}$ greater than or equal to 1.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2* **

ACTION:

With the control banks inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the insertion limit specified in the COLR within 2 hours, or

b. Reduce THERMAL POWER within 2 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

4.1.3.6 The position of each control bank shall be determined to be within the insertion limit specified in the COLR at the frequency specified in the Surveillance Frequency Control Program 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 Specifications 3.10.2 and 3.10.3.

**With $K_{eff}$ greater than or equal to 1.

FIGURE 3.1-2

ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER, THREE LOOP OPERATION

This figure is deleted from Technical Specifications, and is controlled by the CORE OPERATING LIMITS REPORT.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the limits specified in the COLR, either:

1. Restore the indicated AFD to within the limits specified in the COLR within 15 minutes, or

2. 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 less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

* See Special Test Exception 3.10.2

## SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at the frequency specified in the Surveillance Frequency Control Program when the AFD Monitor Alarm is OPERABLE, and

b. Monitoring the indicated AFD for each OPERABLE excore channel at least once within 1 hour and every 1 hour thereafter, when the AFD Monitor Alarm is inoperable.

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

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FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT.

## 3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR – $F_Q(X,Y,Z)$

### LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q^M(X,Y,Z)$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

a. With specification 4.2.2.2.c.1 not being satisfied ($F_Q^M(X,Y,Z)$ exceeding its steady-state limit):

1. Reduce THERMAL POWER ≥ 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 15 minutes.
2. Reduce the Power Range Neutron Flux-High Trip setpoints by ≥ 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 72 hours.
3. Reduce the Overpower ΔT trip setpoints by ≥ 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 72 hours.
4. Prior to increasing THERMAL POWER above the maximum allowable power level from action 3.2.2.a.1, demonstrate through incore flux mapping that $F_Q(X,Y,Z)$ is within its steady-state limit..
5. If the required Actions and associated completion times are not met, be in MODE 2 within 6 hours.

b. With specification 4.2.2.2.c.2 not being satisfied ($F_Q^M(X,Y,Z)$ exceeding its transient Operational limit, $F_Q^L(X,Y,Z)^{OP}$):

1. Reduce AFD limits by the amount specified in the COLR to restore $F_Q(X,Y,Z)$ to within its limits within 4 hours.
2. Reduce THERMAL POWER by the amount specified in the COLR to restore $F_Q(X,Y,Z)$ to within its limits within 4 hours.
3. Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% that the THERMAL POWER level is reduced within 72 hours.
4. Reduce the Overpower ΔT trip setpoints by ≥ 1% for each 1% that the THERMAL POWER level is reduced within 72 hours.
5. Prior to increasing THERMAL POWER above the maximum allowable power level from action 3.2.2.b.2, demonstrate through incore flux mapping that $F_Q(X,Y,Z)$ is within its transient operational limit, $F_Q^L(X,Y,Z)^{OP}$.
6. If the required Actions and associated completion times are not met, be in MODE 2 within 6 hours.

c. With specification 4.2.2.2.c.3 not being satisfied ($F_Q^M(X,Y,Z)$ exceeding its transient Reactor Protection System limit, $F_Q^L(X,Y,Z)^{RPS}$):

1. Reduce Overpower ΔT $f_2(\Delta I)$ breakpoints by KSLOPE for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 72 hours.
2. If the required Actions and associated completion times are not met, be in MODE 2 within 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q^M(X,Y,Z)$ shall be evaluated to determine if it is within its limits by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the $F_Q^M(X,Y,Z)$ of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationships:

1. Steady-state Limit:

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{P} K(Z) * K(BU) \text{ for } P > 0.5$$

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{0.5} K(Z) * K(BU) \text{ for } P \le 0.5$$

where $F_Q^M$(X,Y,Z) is the measured $F_Q(X,Y,Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. $F_Q^{RTP}$ is the $F_Q(X,Y,Z)$ limit at RATED THERMAL POWER provided in the COLR. K(Z) is the normalized $F_Q(X,Y,Z)$ as a function of core height and P is the fraction of RATED THERMAL POWER. K(BU) accounts for degradation of thermal conductivity. $F_Q^{RTP}$, K(Z) and K(BU) are specified in the COLR.

2. Transient Operational Limit:

$$F_Q^M(X,Y,Z) \le F_Q^L(X,Y,Z)^{OP}$$

$$F_Q^L(X,Y,Z)^{OP} = F_Q^D(X,Y,Z) * M_Q(X,Y,Z)$$

where $F_Q^L(X,Y,Z)^{OP}$ is the cycle dependent maximum allowable design peaking factor which ensures that the $F_Q(X,Y,Z)$ limit will be preserved for operation within the LCO limits. $F_Q^L(X,Y,Z)^{OP}$ includes allowances for calculational and measurement uncertainties. $F_Q^D(X,Y,Z)$ is the design power distribution for $F_Q(X,Y,Z)$ provided in the COLR. $M_Q(X,Y,Z)$ is the margin remaining in core location X,Y,Z to the LOCA limit in the transient power distribution and is provided in the COLR for normal operating conditions and power escalation testing during startup operations.

3. Transient Reactor Protection System Limit:

$$F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{RPS}$$

$$F_Q^L(X,Y,Z)^{RPS} = F_Q^D(X,Y,Z) * M_C(X,Y,Z)$$

where $F_Q^L(X,Y,Z)^{RPS}$ is the cycle dependent maximum allowable design peaking factor which ensures that the centerline fuel melt limit will be preserved for operation within the LCO limits. $F_Q^L(X,Y,Z)^{RPS}$ includes allowances for calculational and measurement uncertainties. $M_C(X,Y,Z)$ is the margin remaining to the centerline fuel melt limit in core location X,Y,Z from the transient power distribution and is provided in the COLR for normal operating conditions and power escalation testing during startup operations.

d. Measuring $F_Q^M(X,Y,Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last determined,* or
2. At the frequency specified in the Surveillance Frequency Control Program, whichever occurs first.

---

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

e. Extrapolating $F_Q^M(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement.* If $F_Q^M(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates:

$$F_Q^M(X,Y,Z)_{EXTRAPOLATED} \geq F_Q^L(X,Y,Z)^{OP}_{EXTRAPOLATED},$$

and

$$\frac{F_Q^M(X,Y,Z)_{EXTRAPOLATED}}{F_Q^L(X,Y,Z)^{OP}_{EXTRAPOLATED}} > \frac{F_Q^M(X,Y,Z)}{F_Q^L(X,Y,Z)^{OP}}$$

then:

1. Increase $F_Q^M(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{OP}$; or
2. Repeat Surveillance Requirement 4.2.2.2.c.2 prior to the time at which $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{OP}$ is extrapolated to not be met.

f. Extrapolating $F_Q^M(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement.* If $F_Q^M(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates:

$$F_Q^M(X,Y,Z)_{EXTRAPOLATED} \geq F_Q^L(X,Y,Z)^{RPS}_{EXTRAPOLATED},$$

and

$$\frac{F_Q^M(X,Y,Z)_{EXTRAPOLATED}}{F_Q^L(X,Y,Z)^{RPS}_{EXTRAPOLATED}} > \frac{F_Q^M(X,Y,Z)}{F_Q^L(X,Y,Z)^{RPS}}$$

then:

1. Increase $F_Q^M(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{RPS}$; or
2. Repeat Surveillance Requirement 4.2.2.2.c.3 prior to the time at which $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{RPS}$ is extrapolated to not be met.

---

* Extrapolation of $F_Q^M(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions.

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e, and 4.2.2.2f above are not applicable in the core plane regions specified in the BASES.

4.2.2.3 When $F_Q(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(X,Y,Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

(Pages 3/4 2-7c and 3/4 2-7d have been deleted)

FIGURE 3.2-2

K(Z) - THE NORMALIZED $F_Q(X,Y,Z)$ AS A FUNCTION OF CORE HEIGHT

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT.

## 3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

### LIMITING CONDITION FOR OPERATION

3.2.3 $F^M_{\Delta H}(X,Y)$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

a. With $F^M_{\Delta H}(X,Y)$ outside the limits given in 3.2.3:

1. Within 2 hours reduce THERMAL POWER ≥ RRH%* from RATED THERMAL POWER for each 1% $F^M_{\Delta H}(X,Y)$ exceeds limit.

2. Within 8 hours either:

   a. Restore $F^M_{\Delta H}(X,Y)$ to within the limit for RATED THERMAL POWER, or

   b. Reduce Power Range Neutron Flux – High trip setpoints ≥ RRH%* for each 1% $F^M_{\Delta H}(X,Y)$ exceeds limit.

3. Within 72 hours either:

   a. Restore $F^M_{\Delta H}(X,Y)$ to within limit for RATED THERMAL POWER, or

   b. Reduce Overtemperature ΔT Trip Setpoints by ≥ TRH* for each 1% $F^M_{\Delta H}(X,Y)$ exceeds limit.

* RRH% and TRH are specified in the COLR.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

4. Within 24 hours of $F_{\Delta H}^{M}(X,Y)$ initially being outside the limits of 3.2.3, verify through incore flux mapping that $F_{\Delta H}^{M}(X,Y)$ is within the limits given in 3.2.3.

5. Subsequent POWER OPERATION may proceed provided that $F_{\Delta H}^{M}(X,Y)$ is demonstrated through incore flux mapping to be within acceptable limits prior to exceeding the following THERMAL POWER levels*:

   a) 50% RATED THERMAL POWER

   b) 75% RATED THERMAL POWER

   c) Within 24 hours of attaining greater than or equal to 95% RATED THERMAL POWER

b. With the requirements of ACTION 3.2.3.a not met, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

* THERMAL POWER does not have to be reduced to comply with this ACTION.

## SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F_{\Delta H}^{M}(X,Y)$ shall be evaluated to determine if it is within its limits by:

a. Verifying $F_{\Delta H}^{M}(X,Y)$ is within the steady state limit.

b. Verifying $F_{\Delta H}^{M}(X,Y)$ is within the transient Surveillance limit, $F_{\Delta H}^{L}(X,Y)^{SURV}$

c. Extrapolating $F_{\Delta H}^{M}(X,Y)$ using at least two measurements to 31 EFPD beyond the most recent measurement.* If $F_{\Delta H}^{M}(X,Y)$ is within limits and the 31 EFPD extrapolation indicates:

$$F_{\Delta H}^{M}(X,Y)_{EXTRAPOLATED} \geq F_{\Delta H}^{L}(X,Y)_{EXTRAPOLATED}^{SURV},$$

and

$$\frac{F_{\Delta H}^{M}(X,Y)_{EXTRAPOLATED}}{F_{\Delta H}^{L}(X,Y)_{EXTRAPOLATED}^{SURV}} > \frac{F_{\Delta H}^{M}(X,Y)}{F_{\Delta H}^{L}(X,Y)^{SURV}}$$

then:

1. Increase $F_{\Delta H}^{M}(X,Y)$ by the appropriate factor specified in the COLR and reverify $F_{\Delta H}^{M}(X,Y) \leq F_{\Delta H}^{L}(X,Y)^{SURV}$; or

2. Repeat Surveillance Requirement 4.2.3.2.b prior to the time at which $F_{\Delta H}^{M}(X,Y) \leq F_{\Delta H}^{L}(X,Y)^{SURV}$ is extrapolated to not be met.*

d. Measuring $F_{\Delta H}^{M}(X,Y)$ according to the following schedule:

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

2. At the frequency specified in the Surveillance Frequency Control Program thereafter.

---

* Extrapolation of $F_{\Delta H}^{M}(X,Y)$ is not required for the initial flux map taken after reaching equilibrium conditions.

Figure 3.2-3 Deleted

## 3/4.2.4 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 less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

   a) 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 1.02 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. 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 less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. 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.

*See Special Test Exceptions Specification 3.10.2.

ACTION (Continued):

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

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02, within 30 minutes;

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 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 less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. 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. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

2. 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 less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

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 at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio at the frequency specified in the Surveillance Frequency Control Program when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at the frequency specified in the Surveillance Frequency Control Program.

## 3/4.2.5 DNB PARAMETERS

### LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System $T_{avg}$ ≤ the limit specified in the COLR, and

b. Pressurizer Pressure ≥ the limit specified in the COLR*, and

c. RCS total flow rate ≥ 290,000 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters not within its specified 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 6 hours.

### SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at the frequency specified in the Surveillance Frequency Control Program.

4.2.5.2 Verify, by precision heat balance, that RCS total flow rate is within its limit at the frequency specified in the Surveillance Frequency Control Program.**

* This limit is not applicable during either a THERMAL POWER Ramp in excess of ±5% RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of ±10% RATED THERMAL POWER.

** Required to be performed within 24 hours after ≥ 95% RATED THERMAL POWER.

## 3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.1 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 Each Reactor Trip System instrumentation channels and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit, specified in the Technical Requirements Manual, at the frequency specified in the Surveillance Frequency Control Program.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 1. | Manual Reactor Trip | 2 | 1 | 2 | 1, 2 | 1 |
| | | 2 | 1 | 2 | 3, 4*, 5* | 9 |
| 2. | Power Range, Neutron Flux | | | | | |
| | a. High Setpoint | 4 | 2 | 3 | 1, 2 | 2 |
| | b. Low Setpoint | 4 | 2 | 3 | 1###, 2 | 2 |
| 3. | Power Range, Neutron Flux High Positive Rate | 4 | 2 | 3 | 1, 2 | 2 |
| 4. | Not Used | N/A | N/A | N/A | N/A | N/A |
| 5. | Intermediate Range, Neutron Flux | 2 | 1 | 2 | 1###, 2 | 3 |
| 6. | Source Range, Neutron Flux | | | | | |
| | a. Startup | 2 | 1 | 2 | 2## | 4 |
| | b. Shutdown | 2 | 1 | 2 | 3, 4, 5 | 5 |
| 7. | Overtemperature ΔT | 3 | 2 | 2 | 1, 2 | 6 |
| 8. | Overpower ΔT | 3 | 2 | 2 | 1, 2 | 6 |
| 9. | Pressurizer Pressure--Low (Above P-7) | 3 | 2 | 2 | 1 | 6 (1) |
| 10. | Pressurizer Pressure--High | 3 | 2 | 2 | 1, 2 | 6 |
| 11. | Pressurizer Water Level--High (Above P-7) | 3 | 2 | 2 | 1 | 6 |

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 12. | Reactor Coolant Flow--Low | | | | | |
| | a. Single Loop (Above P-8) | 3/loop | 2/loop in any operating loop | 2/loop in each operating loop | 1 | 6 |
| | b. Two Loops (Above P-7 and below P-8) | 3/loop | 2/loop in two operating loops | 2/loop in each operating loop | 1 | 13a |
| 13. | Steam Generator Water Level--Low-Low | 3/stm. gen | 2/stm. gen. in any operating stm. gen. | 2/stm. gen. each operating stm. gen. | 1, 2 | 6 (1) |
| 14. | Steam Generator Water Level--Low Coincident With Steam / Feedwater Flow Mismatch | 2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen. | 1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen. | 1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen. | 1, 2 | 6 |
| 15. | Undervoltage--Reactor Coolant Pumps (Above P-7) | 2/pump | 2/train | 2/train | 1 | 6 |

## TABLE 3.3-1 (Continued)

## REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 16. | Underfrequency--Reactor Coolant Pumps (Above P-7) | 2/pump | 2/train | 2/train | 1 | 6 |
| 17. | Turbine Trip (Above P-8) | | | | | |
| | a. Low Fluid Oil Pressure | 3 | 2 | 2 | 1 | 6 |
| | b. Turbine Throttle Valve Closure | 4 | 4 | 1 | 1 | 10 |
| 18. | Safety Injection Input from ESF | 2 | 1 | 2 | 1, 2 | 13 |
| 19. | Reactor Trip System Interlocks | | | | | |
| | a. Intermediate Range Neutron Flux, P-6 | 2 | 1 | 2 | 2## | 7 |
| | b. Low Power Reactor Trips Block, P-7 | | | | | |
| | 1) P-10 Input | 4 | 2 | 3 | 1 | 7 |
| | or | | | | | |
| | 2) P-13 Input | 2 | 1 | 2 | 1 | 7 |
| | c. Power Range Neutron Flux, P-8 | 4 | 2 | 3 | 1 | 7 |
| | d. Power Range Neutron Flux, P-10 | 4 | 2 | 3 | 1, 2 | 7 |
| | e. Turbine Inlet Pressure, P-13 | 2 | 1 | 2 | 1 | 7 |

## TABLE 3.3-1 (Continued)

## REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 20. Reactor Trip Breakers | 2<br>2 | 1<br>1 | 2<br>2 | 1, 2<br>3*, 4*, 5* | 8, 11<br>9 |
| 21. Automatic Trip and Interlock Logic | 2<br>2 | 1<br>1 | 2<br>2 | 1, 2<br>3*, 4*, 5* | 13<br>9 |
| 22. Reactor Trip Bypass Breakers | 2 | 1 | 1 | ** | 12 |

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Whenever Reactor Trip Breakers are to be tested.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) The applicable MODES for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

ACTION STATEMENTS

ACTION 1 - 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 in accordance with the Risk-Informed Completion Time Program or be in HOT STANDBY within the next 6 hours.

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 6 hours or in accordance with the Risk-Informed Completion Time Program,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - a. 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 System breakers, and verify compliance with the shutdown margin requirements of Specification 3.1.1.2 within 1 hour and at least once per 12 hours thereafter.

b. With no channels OPERABLE, open the Reactor Trip System breakers within 1 hour and suspend all operations involving positive reactivity changes. Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.2 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 6 hours or in accordance with the Risk-Informed Completion Time Program, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - 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 System breakers within the next hour.

ACTION 10 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or in accordance with the Risk-Informed Completion Time Program or declare the breaker inoperable and apply ACTION 8. 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.

ACTION 12 - No additional corrective actions are required.

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 13a - 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 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

TABLE 3.3-2 REACTOR TRIP SYSTEM INSTRUMENTATION

RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Requirements Manual.

PAGE 3/4 3-10 HAS BEEN DELETED.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. | N.A. | SFCP(12) | N.A. | 1, 2, 3*, 4*, 5* |
| 2. Power Range, Neutron Flux | | | | | | |
| a. High Setpoint | SFCP | SFCP (2,4), SFCP (3,4), SFCP (4,6), SFCP (4,5) | SFCP | N.A. | N.A. | 1, 2 |
| b. Low Setpoint | SFCP | SFCP (4) | S/U(1) | N.A. | N.A. | 1***, 2 |
| 3. Power Range, Neutron Flux, High Positive Rate | N.A. | SFCP (4) | SFCP | N.A. | N.A. | 1, 2 |
| 4. Not Used | N.A. | N.A. | N.A. | N.A. | N.A. | N.A. |
| 5. Intermediate Range, Neutron Flux | SFCP | SFCP (4,5) | S/U(1) | N.A. | N.A. | 1***, 2 |
| 6. Source Range, Neutron Flux | SFCP | SFCP (4,5) | S/U(1), SFCP(8) | N.A. | N.A. | 2**, 3, 4, 5 |
| 7. Overtemperature ΔT | SFCP | SFCP (11) | SFCP | N.A. | N.A. | 1, 2 |
| 8. Overpower ΔT | SFCP | SFCP | SFCP | N.A. | N.A. | 1, 2 |
| 9. Pressurizer Pressure -- Low | SFCP | SFCP | SFCP | N.A. | N.A. | 1 (16) |
| 10. Pressurizer Pressure -- High | SFCP | SFCP | SFCP | N.A. | N.A. | 1, 2 |

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|
| 11. Pressurizer Water Level--High | SFCP | SFCP | SFCP | N.A. | N.A. | 1 |
| 12. Reactor Coolant Flow--Low | SFCP | SFCP | SFCP | N.A. | N.A. | 1 |
| 13. Steam Generator Water Level--Low-Low | SFCP | SFCP | SFCP(16) | N.A. | N.A. | 1, 2 (16) |
| 14. Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch | SFCP | SFCP | SFCP | N.A. | N.A. | 1, 2 |
| 15. Undervoltage -- Reactor Coolant Pumps | N.A. | SFCP | N.A. | SFCP(9) | N.A. | 1 |
| 16. Underfrequency -- Reactor Coolant Pumps | N.A. | SFCP | N.A. | SFCP(9) | N.A. | 1 |
| 17. Turbine Trip | | | | | | |
| a. Low Fluid Oil Pressure | N.A. | SFCP | N.A. | S/U(1,9) | N.A. | 1 |
| b. Turbine Throttle Valve Closure | N.A. | SFCP | N.A. | S/U(1,9) | N.A. | 1 |
| 18. Safety Injection Input from ESF | N.A. | N.A. | N.A. | SFCP | N.A. | 1, 2 |
| 19. Reactor Trip System Interlocks | | | | | | |
| a. Intermediate Range Neutron Flux, P-6 | N.A. | SFCP(4) | SFCP | N.A. | N.A. | 2** |

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|
| 19. Reactor Trip System Interlocks (Continued) | | | | | | |
| b. Low Power Reactor Trips Block, P-7 | N.A. | SFCP(4) | SFCP | N.A. | N.A. | 1 |
| c. Power Range Neutron Flux, P-8 | N.A. | SFCP(4) | SFCP | N.A. | N.A. | 1 |
| d. Power Range Neutron Flux P-10 | N.A. | SFCP(4) | SFCP | N.A. | N.A. | 1, 2 |
| e. Turbine Inlet Pressure, P-13 | N.A. | SFCP | SFCP | N.A. | N.A. | 1 |
| 20. Reactor Trip Breaker | N.A. | N.A. | N.A. | SFCP(7, 9,10) | N.A. | 1, 2, 3*, 4*, 5* |
| 21. Automatic Trip and Interlock Logic | N.A. | N.A. | N.A. | N.A. | SFCP (7) | 1, 2, 3*, 4*, 5* |
| 22. Reactor Trip Bypass Breaker | N.A. | N.A. | N.A. | SFCP (7, 13)<br>SFCP (14) | N.A. | 1, 2, 3*, 4*, 5* |

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at the frequency specified in the Surveillance Frequency Control Program.

(8) Surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(9) Setpoint verification is not applicable.

(10) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the reactor trip breakers.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(11) CHANNEL CALIBRATION shall include the RTD response time.

(12) Verify that appropriate signals reach the undervoltage and shunt trip relays, for both the main and bypass breakers, from the manual reactor trip switch.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(13) Remote manual shunt trip prior to placing breaker in service.

(14) Automatic undervoltage trip.

(15) Not used.

(16) The MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

## 3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features 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 or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 3.3-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 3.3-1 $Z + R + S \leq TA$

   Where:

   Z = The value from Column Z of Table 3.3-4 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

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

## SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within its limit specified in the Technical Requirements Manual at the frequency specified in the Surveillance Frequency Control Program.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps) | | | | | |
| a. Manual Initiation | 2 | 1 | 2 | 1, 2, 3, 4 | 18 |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14 |
| c. Containment Pressure--High-1 | 3 | 2 | 2 | 1, 2, 3, 4 | 26 |
| d. Pressurizer Pressure--Low | 3 | 2 | 2 | 1, 2, 3# | 19 |
| e. Steam Line Pressure--Low | 3/steam line | 2/steam line in any steam line | 2/steam line | 1, 2, 3# | 19 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 2. Containment Spray | | | | | |
| a. Manual Initiation | 2 | 1 with 2 coincident switches | 2 | 1, 2, 3, 4 | 18 |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14 |
| c. Containment Pressure--High-3 | 4 | 2 | 3 | 1, 2, 3 | 16 |
| 3. Containment Isolation | | | | | |
| a. Phase "A" Isolation | | | | | |
| 1) Manual Initiation | 2 | 1 | 2 | 1, 2, 3, 4 | 18 |
| 2) Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14 |
| 3) Safety Injection | See Item 1. above for all Safety Injection initiating functions and requirements. | | | | |
| b. Phase "B" Isolation | | | | | |
| 1) Manual Containment Spray Initiation | See Item 2.a. above for Manual Containment Spray initiating functions and requirements. | | | | |

## TABLE 3.3-3 (Continued)

## ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 3. Containment Isolation (Continued) | | | | | |
| 2) Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14 |
| 3) Containment Pressure--High-3 | See Item 2.c. above for Containment Pressure High-3 initiating functions and requirements. | | | | |
| c. Containment Ventilation Isolation | | | | | |
| 1) Manual Containment Spray Initiation | See Item 2.a. above for Manual Containment Spray initiating functions and requirements. | | | | |
| 2) Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4, 6** | 17, 25 |
| 3) Safety Injection | See Item 1. above for all Safety Injection initiating functions and requirements. | | | | |
| 4) Containment Radioactivity | | | | | |
| a. Area Monitors (both preentry and normal purges) | 4 | See Table 3.3-6, Item 1a, for initiating functions and requirements. | | | |
| b. Airborne Gaseous Radioactivity | | | | | |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 3. Containment Isolation (Continued) | | | | | |
| (1) RCS Leak Detection (normal purge) | 1 | See Table 3.3-6, Item 1b1, for initiating functions and requirements. | | | |
| (2) Preentry Purge Detector | 1 | See Table 3.3-6, Item 1b2, for initiating functions and requirements. | | | |
| c. Airborne Particulate Radioactivity | | | | | |
| (1) RCS Leak Detection (normal purge) | 1 | See Table 3.3-6, Item 1C1, for initiating functions and requirements. | | | |
| (2) Preentry Purge Detector | 1 | See Table 3.3-6, Item 1C2, for initiating functions and requirements. | | | |
| 5) Manual Phase "A" Isolation | See Item 3.a.1) above for Manual Phase "A" Isolation initiating functions and requirements. | | | | |
| 4. Main Steam Line Isolation | | | | | |
| a. Manual Initiation | | | | | |
| 1) Individual MSIV Closure | 1/steam line | 1/steam line | 1/operating steam line | 1, 2, 3, 4 | 23 |
| 2) System | 2 | 1 | 2 | 1, 2, 3 | 27 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 4. | Main Steam Line Isolation (Continued) | | | | | |
| | b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14 |
| | c. Containment Pressure--High-2 | 3 | 2 | 2 | 1, 2, 3 | 26 |
| | d. Steam Line Pressure--Low | See Item 1.e. above for Steam Line Pressure--Low initiating functions and requirements. | | | | |
| | e. Negative Steam Line Pressure Rate--High | 3/steam line | 2 in any steam line | 2/steam line | 3***, 4*** | 19 |
| 5. | Turbine Trip and Feedwater Isolation | | | | | |
| | a. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2 | 24 |
| | b. Steam Generator Water Level--High-High (P-14) | 4/stm. gen. | 2/stm. gen. in any stm. gen. | 3/stm. gen. in each stm. gen. | 1, 2 | 19 |
| | c. Safety Injection | See Item 1. above for all Safety Injection initiating functions and requirements. | | | | |

## TABLE 3.3-3 (Continued)

## ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 6. Auxiliary Feedwater | | | | | |
| a. Manual Initiation | | | | | |
| 1) Motor-Driven Pumps | 1/pump | 1/pump | 1/pump | 1, 2, 3 | 23 |
| 2) Turbine-Driven Pumps | 2/pump | 1/pump | 2/pump | 1, 2, 3 | 23 |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3 | 21 |
| c. Steam Generator Water Level--Low-Low | | | | | |
| 1) Start Motor-Driven Pumps | 3/stm. gen. | 2/stm. gen. in any stm. gen. | 2/stm. gen. in each stm. gen. | 1, 2, 3 | 19 |
| 2) Start Turbine-Driven Pump | 3/stm. gen. | 2/stm. gen. in any 2 stm. gen. | 2/stm. gen. in each stm. gen. | 1, 2, 3 | 19 |
| d. Safety Injection Start Motor-Driven Pumps | See Item 1. above for all Safety Injection initiating functions and requirements. | | | | |
| e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump | See Item 9. below for Loss of Offsite Power initiating functions and requirements. | | | | |
| f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps | 1/pump | 1/pump | 1/pump | 1, 2 | 15 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 6. Auxiliary Feedwater (Continued) | | | | | |
| g. Steam Line Differential Pressure--High | 3/steam line | 2/steam line twice with any steamline low | 2/steam line | 1, 2, 3 | 26 |
| Coincident With Main Steam Line Isolation (Causes AFW Isolation) | See Item 4. above for all Steam Line Isolation initiating functions and requirements | | | | |
| 7. Safety Injection Switchover to Containment Sump | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14 |
| b. RWST Level--Low-Low | 4 | 2 | 3 | 1, 2, 3, 4 | 16 |
| Coincident With Safety Injection | See Item 1. above for all Safety Injection initiating functions and requirements. | | | | |
| 8. Containment Spray Switchover to Containment Sump | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14 |

## TABLE 3.3-3 (Continued)

## ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 8. Containment Spray Switch-over to Containment Sump (Continued) | | | | | |
| b. RWST--Low Low | See Item 7.b. above for all RWST--Low Low initiating functions and requirements. | | | | |
| Coincident With Containment Spray | See Item 2 above for all Containment Spray initiating functions and requirements. | | | | |
| 9. Loss-of-Offsite Power | | | | | |
| a. 6.9 kV Emergency Bus--Undervoltage Primary | 3/bus | 2/bus | 2/bus | 1, 2, 3, 4 | 15a |
| b. 6.9 kV Emergency Bus--Undervoltage Secondary | 3/bus | 2/bus | 2/bus | 1, 2, 3, 4 | 15a |
| 10. Engineered Safety Features Actuation System Interlocks | | | | | |
| a. Pressurizer Pressure, | | | | | |
| P-11 | 3 | 2 | 2 | 1, 2, 3 | 20 |
| Not P-11 | 3 | 2 | 2 | 1, 2, 3 | 20 |
| b. Low-Low $T_{avg}$, P-12 | 3 | 2 | 2 | 1, 2, 3 | 20 |
| c. Reactor Trip, P-4 | 2 | 2 | 2 | 1, 2, 3 ## | 22 |
| d. Steam Generator Water Level, P-14 | See Item 5.b. above for all P-14 initiating functions and requirements. | | | | |

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

**During CORE ALTERATIONS or movement of irradiated fuel in containment, refer to Specification 3.9.9.

***Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

##The functions of the Reactor Trip, P-4 interlock required to meet the LCO are:

- Trip the main turbine – MODES 1 and 2
- Isolate Main Feedwater with coincident low $T_{avg}$ – MODES 1, 2, and 3
- Prevent reactuation of Safety Injection after a manual reset of Safety Injection – MODES 1, 2, and 3
- Prevent opening of Main Feedwater valves if closed on Safety Injection or Steam Generator Water Level – High High – MODES 1, 2, and 3

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or in accordance with the Risk-Informed Completion Time Program or be in at least 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, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 15a - 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 tripped condition within 1 hour. With less than the minimum channels OPERABLE, operation may proceed provided the minimum number of channels is restored within one hour, otherwise declare the affected diesel generator inoperable. When performing surveillance testing of either primary or secondary undervoltage relays, the redundant emergency bus and associated primary and secondary relays shall be OPERABLE.

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 within 6 hours and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the Containment Purge Makeup and Exhaust Isolation valves are maintained closed while in MODES 1, 2, 3 and 4 (refer to Specification 3.6.1.7). For MODE 6, refer to Specification 3.9.4.

ACTION 18 - 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 in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk-Informed Completion Time Program, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or in accordance with the Risk-Informed Completion Time Program 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 provided the other channel is OPERABLE.

ACTION 22 - 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 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated equipment inoperable and take the appropriate ACTION required in accordance with the specific equipment specification.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 25 - During CORE ALTERATIONS or movement of irradiated fuel within containment, comply with the ACTION statement of Specification 3.9.9.

ACTION 26 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 27 - 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 in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

## TABLE 3.3-4

## ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps) | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| c. Containment Pressure--High-1 | 3.64 | 0.71 | 1.5 | ≤ 3.0 psig | ≤ 3.6 psig |
| d. Pressurizer Pressure--Low | 13.5 | 10.47 | 1.5 | ≥ 1850 psig | ≥ 1838 psig |
| e. Steam Line Pressure--Low | 4.52 | 0.71 | 2.0 | ≥ 601 psig | ≥ 581.5 psig* |
| 2. Containment Spray | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| c. Containment Pressure--High-3 | 3.64 | 0.71 | 1.5 | ≤ 10.0 psig | ≤ 11.0 psig |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 3. Containment Isolation | | | | | |
| a. Phase "A" Isolation | | | | | |
| 1) Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2) Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| 3) Safety Injection | See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values. | | | | |
| b. Phase "B" Isolation | | | | | |
| 1) Manual Containment Spray Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2) Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| 3) Containment Pressure--High-3 | See Item 2.c. above for Containment Pressure High-3 Trip Setpoints and Allowable Values. | | | | |
| c. Containment Ventilation Isolation | | | | | |
| 1) Manual Containment Spray Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2) Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 3. Containment Isolation (Continued) | | | | | |
| 3) Safety Injection | See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values. | | | | |
| 4) Containment Radioactivity | | | | | |
| a) Area Monitors (both preentry and normal purges) | See Table 3.3-6, Item 1.a, for trip setpoint. | | | | |
| b) Airborne Gaseous Radioactivity | | | | | |
| (1) RCS Leak Detection (normal purge) | See Table 3.3-6, Item 1.b.1, for trip setpoint. | | | | |
| (2) Preentry Purge Detector | See Table 3.3-6, Item 1.b.2, for trip setpoint. | | | | |
| c) Airborne Particulate Radioactivity | | | | | |
| (1) RCS Leak Detection (normal purge) | See Table 3.3-6, Item 1.c.1, for trip setpoint. | | | | |
| (2) Preentry Purge Detector | See Table 3.3-6, Item 1.c.2, for trip setpoint. | | | | |
| 5) Manual Phase "A" Isolation | N.A. | N.A. | N.A. | N.A. | N.A. |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 4. Main Steam Line Isolation | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| c. Containment Pressure--High-2 | 3.64 | 0.71 | 1.5 | ≤ 3.0 psig | ≤ 3.6 psig |
| d. Steam Line Pressure--Low | See Item 1.e. above for Steam Line Pressure--Low Trip Setpoints and Allowable Values. | | | | |
| e. Negative Steam Line Pressure Rate--High | 2.3 | 0.5 | 0 | ≤ 100 psi# | ≤ 119.5 psi#** |
| 5. Turbine Trip and Feedwater Isolation | | | | | |
| a. Automatic Actuation Logic Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | | | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|---|---|
| 5. | Turbine Trip and Feedwater Isolation (Continued) | | | | | | |
| | b. | Steam Generator Water Level--High-High (P-14) | 22.0 | 8.15 | 2.0 | ≤ 78.0% of narrow range instrument span. | ≤ 79.5% of narrow range instrument span. |
| | c. | Safety Injection | See Item 1. above for Safety Injection Trip Setpoints and Allowable Values. | | | | |
| 6. | Auxiliary Feedwater | | | | | | |
| | a. | Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| | b. | Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| | c. | Steam Generator Water Level--Low-Low | 25.0 | 17.45 | 2.0 | ≥ 25.0% of narrow range instrument span. | ≥ 23.5% of narrow range instrument span. |
| | d. | Safety Injection Start Motor-Driven Pumps | See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values. | | | | |
| | e. | Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump | See Item 9. below for all Loss-of-Offsite Trip Setpoint and Allowable Values. | | | | |
| | f. | Trip of All Main Feedwater Pumps Start Motor-Driven Pumps | N.A. | N.A. | N.A. | N.A. | N.A. |

## TABLE 3.3-4 (Continued)

## ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 6. Auxiliary Feedwater (Continued) | | | | | |
| g. Steam Line Differential Pressure--High | 5.0 | 0.87 | 3.0 | ≤ 100 psi | ≤ 127.4 psi |
| Coincident With Main Steam Line Isolation (Causes AFW Isolation) | See Item 4. above for Main Steam Line Isolation Trip Setpoints and Allowable Values. | | | | |
| 7. Safety Injection Switchover to Containment Sump | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. RWST Level--Low-Low | N.A. | N.A. | N.A. | ≥ 23.4% | ≥ 20.4% |
| Coincident With Safety Injection | See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values. | | | | |
| 8. Containment Spray Switchover to Containment Sump | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. RWST--Low-Low | See Item 7.b. above for all RWST--Low-Low Trip Setpoints and Allowable Values. | | | | |
| Coincident With Containment Spray | See Item 2. above for all Containment Spray Trip Setpoints and Allowable Values. | | | | |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 9. Loss of Offsite Power | | | | | |
| a. 6.9 kV Emergency Bus Undervoltage – Primary (Loss of Voltage) | N.A. | N.A. | N.A. | ≥ 5454 volts with a ≤ 1.46 second time delay (See NOTES 1,2) | ≥ 5329 volts with a ≤ 1.5 second time delay |
| b. 6.9 kV Emergency Bus Undervoltage – Secondary (Degraded Voltage) | N.A. | N.A. | N.A. | ≥ 6420 volts with a ≤ 12.88 second time delay (with Safety Injection). | ≥ 6392 volts with a ≤ 13.21 second time delay (with Safety Injection). |
| | | | | ≥ 6420 volts with a ≤ 57.89 second time delay (non-accident). | ≥ 6392 volts with a ≤ 59.62 second time delay (non-accident). |
| 10. Engineered Safety Features Actuation System Interlocks | | | | | |
| a. Pressurizer Pressure, | | | | | |
| P-11 | N.A. | N.A. | N.A. | ≥ 2000 psig | ≥ 1988 psig |
| Not P-11 | N.A. | N.A. | N.A. | ≤ 2000 psig | ≤ 2012 psig |
| b. Low Low $T_{avg}$, P-12 | N.A. | N.A. | N.A. | ≥ 553°F | ≥ 549.3°F |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 10. Engineered Safety Features Actuation System Interlocks (Continued) | | | | | |
| c. Reactor Trip, P-4 | N.A. | N.A. | N.A. | N.A. | N.A. |
| d. Steam Generator Water Level, P-14 | See Item 5.b above for all Steam Generator Water Level Trip Setpoints and Allowable Values. | | | | |

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

* Time constants utilized in the lead-lag controller for Steam Line Pressure--Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

** The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate--High is ≥ 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

\# The indicated values are the effective, cumulative, rate-compensated pressure drops as seen by the comparator.

NOTE 1: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 2: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 3.3-4 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in the FSAR. The as-found and as-left tolerances are specified in the Technical Requirements Manual.

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Requirements Manual.

PAGES 3/4 3-38 THROUGH 3/4-40 HAVE BEEN DELETED.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps) | | | | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP | 1, 2, 3, 4 |
| c. Containment Pressure -- High-1 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| d. Pressurizer Pressure -- Low | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| e. Steam Line Pressure -- Low | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 2. Containment Spray | | | | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP | 1, 2, 3. 4 |
| c. Containment Pressure--High-3 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| 3. Containment Isolation | | | | | | | | |
| a. Phase "A" Isolation | | | | | | | | |
| 1) Manual Initiation | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| 2) Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP | 1, 2, 3, 4 |
| 3) Safety Injection | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |
| b. Phase "B" Isolation | | | | | | | | |
| 1) Manual Containment Spray Initiation | See Item 2.a. above for Manual Containment Spray Surveillance Requirements. | | | | | | | |
| 2) Automatic Actuation Logic Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 3. Containment Isolation (Continued) | | | | | | | | |
| 3) Containment Pressure --High-3 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| c. Containment Ventilation Isolation | | | | | | | | |
| 1) Manual Containment Spray Initiation | See Item 2.a. above for Manual Containment Spray Surveillance Requirements. | | | | | | | |
| 2) Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1, 2) | SFCP(1, 2) | SFCP(2) | 1, 2, 3, 4, 6# |
| 3) Safety Injection | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |
| 4) Containment Radioactivity | | | | | | | | |
| a) Area Monitors (both preentry and normal purges) | See Table 4.3-3, Item 1a, for surveillance requirements. | | | | | | | |
| b) Airborne Gaseous Radioactivity | | | | | | | | |
| (1) RCS Leak Detection (normal purge) | See Table 4.3-3, Item 1b1, for surveillance requirements. | | | | | | | |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 3. Containment Isolation (Continued) | | | | | | | | |
| (2) Preentry Purge Detector | See Table 4.3-3, Item 1b2, for surveillance requirements. | | | | | | | |
| c) Airborne Particulate Radioactivity | | | | | | | | |
| (1) RCS Leak Detection (normal purge) | See Table 4.3-3, Item 1C1, for surveillance requirements. | | | | | | | |
| (2) Preentry Purge Detector | See Table 4.3-3, Item 1C2, for surveillance requirements. | | | | | | | |
| 5) Manual Phase A Isolation | See Item 3.a.1) above for Manual Phase A Isolation Surveillance Requirements. | | | | | | | |
| 4. Main Steam Line Isolation | | | | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1)(4) | SFCP(1) | SFCP | 1, 2, 3, 4 |
| c. Containment Pressure--High-2 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 4. Main Steam Line Isolation (Continued) | | | | | | | | |
| d. Steam Line Pressure --Low | See Item 1.e. above for Steam Line Pressure --Low Surveillance Requirements. | | | | | | | |
| e. Negative Steam Line Pressure Rate--High | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 3**, 4** |
| 5. Turbine Trip and Feedwater Isolation | | | | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP | 1, 2 |
| b. Steam Generator Water Level--High -High (P-14) | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2 |
| c. Safety Injection | See Item 1. above for Safety Injection Surveillance Requirements. | | | | | | | |
| 6. Auxiliary Feedwater | | | | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3 |
| b. Automatic Actuation and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP | 1, 2, 3 |
| c. Steam Generator Water Level--Low-Low | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| d. Safety Injection Start Motor-Driven Pumps | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |
| e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump | See Item 9. below for all Loss-of-Offsite Power Surveillance Requirements. | | | | | | | |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 6. Auxiliary Feedwater (Continued) | | | | | | | | |
| f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2 |
| g. Steam Line Differential Pressure--High | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | SFCP | 1, 2, 3 |
| Coincident With Main Steam Line Isolation (Causes AFW Isolation) | See Item 4. above for all Main Steam Line Isolation Surveillance Requirements. | | | | | | | |
| 7. Safety Injection Switchover to Containment Sump | | | | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP | 1, 2, 3, 4 |
| b. RWST Level --Low-Low | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | SFCP | 1, 2, 3, 4 |
| Coincident With Safety Injection | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |
| 8. Containment Spray Switchover to Containment Sump | | | | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 8. Containment Spray Switchover to Containment Sump (Continued) | | | | | | | | |
| b. RWST Level--Low-Low | See Item 7.b. above for RWST Level--Low-Low Surveillance Requirements. | | | | | | | |
| Coincident with Containment Spray | See Item 2. above for Containment Spray Surveillance Requirements. | | | | | | | |
| 9. Loss-of-Offsite Power | | | | | | | | |
| a. 6.9 kV Emergency Bus Undervoltage--Primary | N.A. | SFCP | N.A. | SFCP* | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| b. 6.9 kV Emergency Bus Undervoltage --Secondary | N.A. | SFCP | N.A. | SFCP* | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| 10. Engineered Safety Features Actuation System Interlocks | | | | | | | | |
| a. Pressurizer Pressure, | | | | | | | | |
| P-11 | N.A. | SFCP | SFCP | N.A. | SFCP(1) | SFCP(1) | N.A. | 1, 2, 3 |
| Not P-11 | N.A. | SFCP | SFCP | N.A. | SFCP(1) | SFCP(1) | N.A. | 1, 2, 3 |
| b. Low-Low $T_{avg}$, P-12 | N.A. | SFCP | SFCP | N.A. | SFCP(1) | SFCP(1) | N.A. | 1, 2, 3 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 10. Engineered Safety Features Actuation System Interlocks (Continued) | | | | | | | | |
| c. Reactor Trip, P-4 | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3 ## |
| d. Steam Generator Water Level, P-14 | See Item 5.b., above for P-14 Surveillance Requirements. | | | | | | | |

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at the frequency specified in the Surveillance Frequency Control Program.

(2) The Surveillance Requirements of Specification 4.9.9 apply during CORE ALTERATIONS or movement of irradiated fuel in containment.

(3) Deleted.

(4) The Steam Line Isolation-Safety Injection (Block-Reset) switches enable the Negative Steam Line Pressure Rate--High signal (item 4.e) when used below the P-11 setpoint. Verify proper operation of these switches each time they are used.

* Setpoint verification not required.

\# During CORE ALTERATIONS or movement of irradiated fuel in containment.

** Trip Function automatically blocked above P-11 and may be blocked below P-11 when safety injection or low steamline pressure is not blocked.

\#\# The functions of the Reactor Trip, P-4 interlock required to meet the LCO are:

- Trip the main turbine – MODES 1 and 2
- Isolate Main Feedwater with coincident low $T_{avg}$ – MODES 1, 2, and 3
- Prevent reactuation of Safety Injection after a manual reset of Safety Injection – MODES 1, 2, and 3
- Prevent opening of Main Feedwater valves if closed on Safety Injection or Steam Generator Water Level – High High – MODES 1, 2, and 3

## 3/4.3.3 MONITORING INSTRUMENTATION

### RADIATION MONITORING FOR PLANT OPERATIONS

### LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations 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 for plant operations 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 for plant operations 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 for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

## TABLE 3.3-6

## RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

| INSTRUMENT | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM/TRIP SETPOINT | ACTION |
|---|---|---|---|---|---|
| 1. Containment Radioactivity-- | | | | | |
| a. Containment Ventilation Isolation Signal Area Monitors | 2 | 3 | 1, 2, 3, 4, 6 | # | 27 |
| b. Airborne Gaseous Radioactivity | | | | | |
| 1) RCS Leakage Detection | 1 | 1 | 1, 2, 3, 4 | $\leq 1.0 \times 10^{-3}$ µCi/ml | 26, 27 |
| 2) Pre-entry Purge | 1 | 1 | ## | $\leq 2.0 \times 10^{-3}$ µCi/ml | 30 |
| c. Airborne Particulate Radioactivity | | | | | |
| 1) RCS Leakage Detection | 1 | 1 | 1, 2, 3, 4 | $\leq 4.0 \times 10^{-8}$ µCi/ml | 26, 27 |
| 2) Pre-entry Purge | 1 | 1 | ## | $\leq 1.5 \times 10^{-8}$ µCi/ml | 30 |
| 2. Spent Fuel Pool Area-- Fuel Handling Building Emergency Exhaust Actuation | | | | | |
| a. Fuel Handling Building Operating Floor--South Network | 1/train*** | 1/train 2 trains | ** | ≤ 100 mR/hr | 28 |
| b. Fuel Handling Building Operating Floor--North Network | 1/train*** | 1/train 2 trains | * | ≤ 100 mR/hr | 28 |
| 3. Control Room Outside Air Intakes-- | | | | | |
| a. Normal Outside Air Intake Isolation | 1 | 2 | 1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools. | $\leq 4.9 \times 10^{-6}$ µCi/ml | 29 |

## TABLE 3.3-6 (Continued)

## RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

| INSTRUMENT | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM/TRIP SETPOINT | ACTION |
|---|---|---|---|---|---|
| 3. Control Room Outside Air Intakes-- (Continued) | | | | | |
| b. Emergency Outside Air Intake Isolation--South Intake | 1 | 2 | 1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools. | $\leq 4.9 \times 10^{-6}$ µCi/ml | 29 |
| c. Emergency Outside Air Intake Isolation--North Intake | 1 | 2 | 1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools. | $\leq 4.9 \times 10^{-6}$ µCi/ml | 29 |

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* With irradiated fuel in the Northend Spent Fuel Pool or transfer of irradiated fuel from or to a spent fuel shipping cask.

** With irradiated fuel in the Southend Spent Fuel Pool or New Fuel Pool.

*** Each channel consists of 3 detectors with 1 of 3 logic. A channel is OPERABLE when 1 or more of the detectors are OPERABLE.

\# For MODES 1, 2, 3 and 4, the setpoint shall be less than or equal to three times detector background at RATED THERMAL POWER. During fuel movement the setpoint shall be less than or equal to 150 mR/hr.

\#\# Required OPERABLE whenever pre-entry purge system is to be used.

ACTION STATEMENTS

ACTION 26 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 27 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge makeup and exhaust isolation valves are maintained closed.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, declare the associated train of Fuel Handling Building Emergency Exhaust inoperable and perform the requirements of Specification 3.9.12.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour initiate isolation of the respective air intake. With no outside air intakes available, maintain operation of the Control Room Emergency Filtration System in the Recirculation Mode of Operation.

ACTION 30 - With less than the Minimum Channels OPERABLE requirement, pre-entry purge operations shall be suspended and the containment pre-entry purge makeup and exhaust valves shall be maintained closed.

## TABLE 4.3-3

## RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION | DIGITAL CHANNEL OPERATIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| 1. Containment Radioactivity-- | | | | |
| a. Containment Ventilation Isolation Signal Area Monitors | SFCP | SFCP | SFCP | 1, 2, 3, 4, 6 |
| b. Airborne Gaseous Radioactivity | | | | |
| 1) RCS Leakage Detection | SFCP | SFCP | SFCP | 1, 2, 3, 4 |
| 2) Pre-entry Purge | SFCP | SFCP | SFCP## | # |
| c. Airborne Particulate Radioactivity | | | | |
| 1) RCS Leakage Detection | SFCP | SFCP | SFCP | 1, 2, 3, 4 |
| 2) Pre-entry Purge | SFCP | SFCP | SFCP## | # |
| 2. Spent Fuel Pool Area -- Fuel Handling Building Emergency Exhaust Actuation | | | | |
| a. Fuel Handling Building Operating Floor--South Network | SFCP | SFCP | SFCP | ** |
| b. Fuel Handling Building Operating Floor--North Network | SFCP | SFCP | SFCP | * |

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION | DIGITAL CHANNEL OPERATIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| 3. Control Room Outside Air Intakes | | | | |
| a. Normal Outside Air Intake Isolation | SFCP | SFCP | SFCP | 1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools. |
| b. Emergency Outside Air Intake Isolation--South Intake | SFCP | SFCP | SFCP | 1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools. |
| c. Emergency Outside Air Intake Isolation--North Intake | SFCP | SFCP | SFCP | 1,2,3,4,5,6 and during movement of irradiated fuel assemblies and movement of loads over spent fuel pools. |

TABLE NOTATIONS

* With irradiated fuel in the Northend Spent Fuel Pool or transfer of irradiated fuel from or to a spent fuel shipping cask.

** With irradiated fuel in the Southend Spent Fuel Pool or New Fuel Pool.

\# Whenever pre-entry purge system is to be used.

\#\# Prior to operation of pre-entry purge unless performed within the last 92 days.

MOVABLE INCORE DETECTORS - DELETED

## LIMITING CONDITION FOR OPERATION

3.3.3.3 Deleted

TABLE 3.3-7 Deleted

TABLE 4.3-4 Deleted

## LIMITING CONDITION FOR OPERATION

3.3.3.4 Deleted

TABLE 3.3-8 Deleted

TABLE 4.3-5 Deleted

## LIMITING CONDITION FOR OPERATION

3.3.3.5.a The Remote Shutdown System monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

3.3.3.5.b All transfer switches, Auxiliary Control Panel Controls and Auxiliary Transfer Panel Controls for the OPERABILITY of those components required by the SHNPP Safe Shutdown Analysis to (1) remove decay heat via auxiliary feedwater flow and steam generator power-operated relief valve flow from steam generators A and B, (2) control RCS inventory through the normal charging flow path, (3) control RCS pressure, (4) control reactivity, and (5) remove decay heat via the RHR system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown monitoring channels less than the Total Number of Channels required by Table 3.3-9, restore the inoperable channels to OPERABLE status within 60 days or submit a Special Report in accordance with Specification 6.9.2 within 14 additional days.

c. With one or more inoperable Remote Shutdown System transfer switches, power, or control circuits required by 3.3.3.5.b, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

## SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit and control switch required by 3.3.3.5.b, shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program.

## TABLE 3.3-9

## REMOTE SHUTDOWN SYSTEM

| INSTRUMENT | READOUT LOCATION | TOTAL NO. OF CHANNELS | MINIMUM CHANNELS OPERABLE |
|---|---|---|---|
| 1. Reactor Coolant System Hot-Leg Temperature | ACP* | 2 | 2 |
| 2. Reactor Coolant System Cold-Leg Temperature | ACP* | 2 | 2 |
| 3. Pressurizer Pressure | ACP* | 2 | 1-SSA Channel** |
| 4. Pressurizer Level | ACP* | 2 | 1-SSA Channel** |
| 5. Steam Generator Pressure (Note 1) | ACP* | 1/Steam Generator | 1/Steam Generator |
| 6. Steam Generator Water Level--Wide Range (Note 1) | ACP* | 1/Steam Generator | 1/Steam Generator |
| 7. Residual Heat Removal Flow | ACP* | 2 | 1 (Note 2) |
| 8. Auxiliary Feedwater Flow (Note 1) | ACP* | 1/Steam Generator | N.A. (Note 3) |
| 9. Condensate Storage Tank Level | ACP* | 2 | 1-SSA Channel** |
| 10. Reactor Coolant System Pressure-Wide Range | ACP* | 2 | 1-SSA Channel** |
| 11. Wide-Range Flux Monitor (SR Indicator) | ACP* | 1 | 1-SSA Channel** |
| 12. Charging Header Flow | ACP* | 1 | 1-SSA Channel** |
| 13. a. Auxiliary Feedwater Turbine Steam Inlet--Pump Discharge ΔP or | ACP* | 1 | 1-SSA Channel** |
| b. Auxiliary Feedwater Turbine Speed | ACP* | 1 | 1-SSA Channel** |
| 14. Boric Acid Tank Level | ACP* | 1 | 1-SSA Channel** |

*ACP = Auxiliary Control Panel
**SSA = Safe Shutdown Analysis

Note 1 - Steam Generators A&B Only
Note 2 - RHR Train B Only
Note 3 - Steam Generator Water Level is used

## TABLE 4.3-6

## REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|---|
| 1. | Reactor Coolant System Hot-Leg Temperature | SFCP | SFCP |
| 2. | Reactor Coolant System Cold-Leg Temperature | SFCP | SFCP |
| 3. | Pressurizer Pressure | SFCP | SFCP |
| 4. | Pressurizer Level | SFCP | SFCP |
| 5. | Steam Generator Pressure | SFCP | SFCP |
| 6. | Steam Generator Water Level--Wide Range | SFCP | SFCP |
| 7. | Residual Heat Removal Flow | SFCP | SFCP |
| 8. | Auxiliary Feedwater Flow | SFCP | SFCP |
| 9. | Condensate Storage Tank Level | SFCP | SFCP |
| 10. | Reactor Coolant System Pressure--Wide Range | SFCP | SFCP |
| 11. | Wide-Range Flux Monitor (SR Indicator) | SFCP | SFCP |
| 12. | Charging Header Flow | SFCP | SFCP |
| 13. | a. Auxiliary Feedwater Turbine Steam Inlet--Pump Discharge $\Delta P$ | SFCP | SFCP |
| | b. Auxiliary Feedwater Turbine Speed | SFCP | SFCP |
| 14. | Boric Acid Tank Level | SFCP | SFCP |

## LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

------------------------------------------------------------NOTE------------------------------------------------------------

A separate ACTION entry is allowed for each INSTRUMENT listed in Table 3.3-10.

--------------------------------------------------------------------------------------------------------------------------------

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Required Number of Channels requirements shown in Table 3.3-10 restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report pursuant to Specification 6.9.2 within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel(s) to operable status.

b. With the number of OPERABLE accident monitoring instrumentation channels, except the radiation monitors, the Pressurizer Safety Valve Position Indicator, or the Reactor Coolant System Subcooling Margin Monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE accident monitoring instrumentation channels for the radiation monitor(s), the Pressurizer Safety Valve Position Indicator*, or the Reactor Coolant System Subcooling Margin Monitor#, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within the next 14 days, that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channel(s) to OPERABLE status.

d. DELETED.

e. DELETED.

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* The alternate method shall be a check of safety valve piping temperatures and evaluation to determine position.

\# The alternate method shall be the initiation of the backup method as required by Specification 6.8.4.d.

## SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

| INSTRUMENT | TOTAL REQUIRED NO. OF CHANNELS | MINIMUM CHANNELS OPERABLE |
|---|---|---|
| 1. Containment Pressure | | |
| a. Narrow Range | 2 | 1 |
| b. Wide Range | 2 | 1 |
| 2. Reactor Coolant Hot-Leg Temperature--Wide Range | 2 | 1 |
| 3. Reactor Coolant Cold-Leg Temperature--Wide Range | 2 | 1 |
| 4. Reactor Coolant Pressure--Wide Range | 2 | 1 |
| 5. Pressurizer Water Level | 2 | 1 |
| 6. Steam Line Pressure | 2/steam generator | 1/steam generator |
| 7. Steam Generator Water Level--Narrow Range | N.A. | 1/steam generator |
| 8. Steam Generator Water Level--Wide Range | N.A. | 1/steam generator |
| 9. Refueling Water Storage Tank Water Level | 2 | 1 |
| 10. Auxiliary Feedwater Flow Rate | N.A. | 1/steam generator |
| 11. Reactor Coolant System Subcooling Margin Monitor | N.A. | 1 |
| 12. PORV Position Indicator* | N.A. | 1/valve |
| 13. PORV Block Valve Position Indicator** | N.A. | 1/valve |
| 14. Pressurizer Safety Valve Position Indicator | N.A. | 1/valve |
| 15. Containment Water Level (ECCS Sump)--Narrow Range | 2 | 1 |
| 16. Containment Water Level--Wide Range | 2 | 1 |

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

| INSTRUMENT | TOTAL REQUIRED NO. OF CHANNELS | MINIMUM CHANNELS OPERABLE |
|---|---|---|
| 17. In Core Thermocouples | 4/core quadrant | 2/core quadrant |
| 18. Plant Vent Stack--High Range Noble Gas Radiation Monitor | N.A. | 1 |
| 19. Main Steam Line Radiation Monitors | N.A. | 1/steam line |
| 20. Containment--High Range Radiation Monitor | N.A. | 1 |
| 21. Reactor Vessel Level | 2 | 1 |
| 22. Containment Spray NaOH Tank Level | 2 | 1 |
| 23. Turbine Building Vent Stack High Range Noble Gas Radiation Monitor | N.A. | 1 |
| 24. Waste Processing Building Vent Stack High Range Noble Gas Radiation Monitors | | |
| a. Vent Stack 5 | N.A. | 1 |
| b. Vent Stack 5A | N.A. | 1 |
| 25. Condensate Storage Tank Level | 2 | 1 |

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|---|
| 1. | Containment Pressure | | |
| | a. Narrow Range | SFCP | SFCP |
| | b. Wide Range | SFCP | SFCP |
| 2. | Reactor Coolant Hot-Leg Temperature--Wide Range | SFCP | SFCP |
| 3. | Reactor Coolant Cold-Leg Temperature--Wide Range | SFCP | SFCP |
| 4. | Reactor Coolant Pressure--Wide Range | SFCP | SFCP |
| 5. | Pressurizer Water Level | SFCP | SFCP |
| 6. | Steam Line Pressure | SFCP | SFCP |
| 7. | Steam Generator Water Level--Narrow Range | SFCP | SFCP |
| 8. | Steam Generator Water Level--Wide Range | SFCP | SFCP |
| 9. | Refueling Water Storage Tank Water Level | SFCP | SFCP |
| 10. | Auxiliary Feedwater Flow Rate | SFCP | SFCP |
| 11. | Reactor Coolant System Subcooling Margin Monitor | SFCP | SFCP |
| 12. | PORV Position Indicator | SFCP | SFCP |
| 13. | PORV Block Valve Position Indicator | SFCP | SFCP |
| 14. | Pressurizer Safety Valve Position Indicator | SFCP | SFCP |
| 15. | Containment Water Level (ECCS Sump)--Narrow Range | SFCP | SFCP |
| 16. | Containment Water Level--Wide Range | SFCP | SFCP |

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|---|
| 17. | In Core Thermocouples | SFCP | SFCP |
| 18. | Plant Vent Stack--High Range Noble Gas Monitor | SFCP | SFCP |
| 19. | Main Steam Line Radiation Monitors | SFCP | SFCP |
| 20. | Containment--High Range Radiation Monitor | SFCP | SFCP* |
| 21. | Reactor Vessel Level | SFCP | SFCP |
| 22. | Containment Spray NaOH Tank Level | SFCP | SFCP |
| 23. | Turbine Building Vent Stack High Range Noble Gas Monitor | SFCP | SFCP |
| 24. | Waste Processing Building Vent Stack High Range Noble Gas Monitors | | |
| | a. Vent Stack 5 | SFCP | SFCP |
| | b. Vent Stack 5A | SFCP | SFCP |
| 25. | Condensate Storage Tank Level | SFCP | SFCP |

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

## CHLORINE DETECTION SYSTEMS

Specification 3/4 3.3.7 deleted.

## FIRE DETECTION INSTRUMENTATION

Specification 3/4 3.3.8 DELETED
Table 3.3-11 DELETED

LIMITING CONDITION FOR OPERATION

3.3.3.9 Deleted

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Specification 3/4.3.3.10 has been deleted from Technical Specifications and has been relocated to the ODCM.

Pages 3/4 3-76 through 3/4 3-81 have been deleted.

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 Deleted

TABLE 3.3-13 Deleted

Pages 3/4 3-84 and 3/4 3-85 have been deleted.

TABLE 4.3-9 Deleted

Pages 3/4 3-87 and 3/4 3-88 have been deleted.

3/4.3.4 TURBINE OVERSPEED PROTECTION - DELETED

## 3/4.4 REACTOR COOLANT SYSTEM

### 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

### STARTUP AND POWER 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 6 hours.

#### SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at the frequency specified in the Surveillance Frequency Control Program.

*See Special Test Exceptions Specification 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

## LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant pumps in operation when the Reactor Trip System breakers are closed or with one reactor coolant pump in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

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 only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

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

## SURVEILLANCE REQUIREMENTS

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

*All reactor coolant pumps may be deenergized 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 (CONTINUED)

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying narrow range secondary side water level to be greater than or equal to 30% at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at the frequency specified in the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. RHR Loop A, or

e. RHR Loop B.

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; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no 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 loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized 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.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying wide range (WR) secondary side water level is greater than 74% or narrow range (NR) secondary side water level is greater than 30% at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at the frequency specified in the Surveillance Frequency Control Program .

4.4.1.3.4 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.*

* Not required to be performed until 12 hours after entering Mode 4.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

## LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than 74% wide range (WR) or greater than 30% narrow range (NR).

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level 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

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.4.1.3 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

---

* The RHR pump may be deenergized 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.

** One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

*** A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

## LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR 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

4.4.1.4.2.1 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.4.2.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

---

* One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

** The RHR pump may be deenergized 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.

REACTOR COOLANT SYSTEM

3/4.4.2 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 ± 1%.*

APPLICABILITY: MODES 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 requirements other than those required by the INSERVICE TESTING PROGRAM.

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

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

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 at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

## 3/4.4.3 PRESSURIZER

### LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water level of less than or equal to 75% of indicated span, and at least two groups of pressurizer heaters each having a capacity of at least 125 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups 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 the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water level shall be determined to be within its limit at the frequency specified in the Surveillance Frequency Control Program.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit power (kW) at the frequency specified in the Surveillance Frequency Control Program.

## 3/4.4.4 RELIEF VALVES

### LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one or more PORV(s) 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 or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and

    1. With only one safety grade PORV OPERABLE, restore at least a total of two safety grade PORVs to OPERABLE status within the following 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

    2. With no safety grade PORVs OPERABLE, restore at least one safety grade PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable safety grade PORV or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply ACTION b.1 or b.2, above, as appropriate, for the isolated PORV(s).

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of the INSERVICE TESTING PROGRAM, each PORV shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Performing a CHANNEL CALIBRATION of the actuation instrumentation, and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4, prior to going to 325°F.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The accumulator for the safety-related PORVs shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by isolating the normal air and nitrogen supplies and operating the valves through a complete cycle of full travel.

## 3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

All steam generator 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. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, 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.

   AND

b. With the requirements and associated allowed outage time of ACTION a., above, not met or SG tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

### SURVEILLANCE REQUIREMENTS

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

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

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* Separate ACTION entry is allowed for each SG tube.

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Table 4.4-1
Deleted by Amendment

Table 4.4-2
Deleted by Amendment

TABLE 4.4-2B Deleted

TABLE 4.4-2C Deleted

## 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 Airborne Gaseous Radioactivity Monitoring System,
b. The Reactor Cavity Sump Level and Flow Monitoring System, and
c. The Containment Airborne Particulate Radioactivity Monitoring System.

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

ACTION:

a. With a. or c. of the above required Leakage Detection Systems INOPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for airborne gaseous and particulate radioactivity at least once per 24 hours when the required Airborne Gaseous 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.

b. With b. of the above required Leakage Detection Systems inoperable be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a. and c. of the above required Leakage Detection Systems inoperable:

   1. Restore either Monitoring System (a. or c.) to OPERABLE status within 72 hours and

   2. Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours, and

   3. Perform a Reactor Coolant System water inventory balance at least once per 8 hours*.

   Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---

*Not required to be performed until 12 hours after establishment of steady-state operation. Not applicable to primary-to-secondary leakage.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. For Containment Airborne Gaseous and Particulate Monitoring Systems, performance of CHANNEL CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3.

b. For Reactor Cavity Sump Level and Flow Monitoring System, performance of CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational 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,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 31 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 ± 20 psig.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding primary to secondary leakage, PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, 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.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, 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.

d. With 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.

* Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted by multiplying the observed leakage by the square root of the quotient of 2235 divided by the test pressure.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated, at the frequency specified in the Surveillance Frequency Control Program, to be within each of the above limits by:

a. Monitoring the containment Airborne Gaseous or Particulate Radioactivity Monitor;

b. Monitoring the containment sump inventory and Flow Monitoring System;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance*; and

e. Monitoring the Reactor Head Flange Leakoff System.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. In accordance with the INSERVICE TESTING PROGRAM.

b. (DELETED)

c. (DELETED)

d. (DELETED)

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

4.4.6.2.3 Primary-to-secondary leakage shall be verified to be ≤ 150 gallons per day through any one steam generator at the frequency specified in the Surveillance Frequency Control Program**.

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* Not required to be performed until 12 hours after establishment of steady-state operation. Not applicable to primary-to-secondary leakage.

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

| EBASCO VALVE NUMBER | CP&L VALVE NUMBER | TYPE | FUNCTION | MAXIMUM ALLOWABLE LEAKAGE |
|---|---|---|---|---|
| 1-RH-V502-SB-1 | 1RH1 | 12" Gate | RHR Pump Suction | 5 gpm |
| 1-RH-V503-SA-1 | 1RH2 | 12" Gate | RHR Pump Suction | 5 gpm |
| 1-RH-V500-SB-1 | 1RH39 | 12" Gate | RHR Pump Suction | 5 gpm |
| 1-RH-V501-SA-1 | 1RH40 | 12" Gate | RHR Pump Suction | 5 gpm |
| 1-SI-V510-SA-1 | 1SI134 | 6" Check | Low Head Injection (Hot Leg) | 3 gpm |
| 1-SI-V511-SB-1 | 1SI135 | 6" Check | Low Head Injection (Hot Leg) | 3 gpm |
| 1-SI-V544-SA-1 | 1SI249 | 12" Check | Accumulator Injection | 5 gpm |
| 1-SI-V547-SA-1 | 1SI250 | 12" Check | Accumulator Injection | 5 gpm |
| 1-SI-V545-SB-1 | 1SI251 | 12" Check | Accumulator Injection | 5 gpm |
| 1-SI-V548-SB-1 | 1SI252 | 12" Check | Accumulator Injection | 5 gpm |
| 1-SI-V546-SA-1 | 1SI253 | 12" Check | Accumulator Injection | 5 gpm |
| 1-SI-V549-SA-1 | 1SI254 | 12" Check | Accumulator Injection | 5 gpm |
| 2-SI-V581-SA-1 | 1SI346 | 10" Check | Low Head Injection | 5 gpm |
| 2-SI-V580-SB-1 | 1SI347 | 10" Check | Low Head Injection | 5 gpm |
| 1-SI-V584-SA-1 | 1SI356 | 6" Check | Low Head Injection | 3 gpm |
| 1-SI-V585-SB-1 | 1SI357 | 6" Check | Low Head Injection | 3 gpm |
| 1-SI-V586-SA-1 | 1SI358 | 6" Check | Low Head Injection | 3 gpm |
| 1-SI-V587-SA-1 | 1SI359 | 10" Gate | Hot Leg Recirculation | 5 gpm |

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

## LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

## SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

## TABLE 3.4-2

## REACTOR COOLANT SYSTEM

## CHEMISTRY LIMITS

| PARAMETER | STEADY-STATE LIMIT | TRANSIENT LIMIT |
|---|---|---|
| Dissolved Oxygen* | $\leq$ 0.10 ppm | $\leq$ 1.00 ppm |
| Chloride | $\leq$ 0.15 ppm | $\leq$ 1.50 ppm |
| Fluoride | $\leq$ 0.15 ppm | $\leq$ 1.50 ppm |

*Limit not applicable with $T_{avg}$ less than or equal to 250°F.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

| PARAMETER | SAMPLE AND ANALYSIS FREQUENCY |
| --- | --- |
| Dissolved Oxygen* | At the frequency specified in the Surveillance Frequency Control Program |
| Chloride | At the frequency specified in the Surveillance Frequency Control Program |
| Fluoride | At the frequency specified in the Surveillance Frequency Control Program |

* Not required with $T_{avg}$ less than or equal to 250°F

## 3/4.4.8 SPECIFIC ACTIVITY

### LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 60.0 microCurie per gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY with $T_{avg}$ less than 500°F within 6 hours. LCO 3.0.4.c is applicable to DOSE EQUIVALENT I-131.

b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with $T_{avg}$ less than 500°F within 6 hours.

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

With the specific activity of the reactor coolant greater than 1 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

### SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

* With $T_{avg}$ greater than or equal to 500°F.

FIGURE 3.4-1 Deleted

## TABLE 4.4-4

## REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY | MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED |
|---|---|---|
| 1. Gross Radioactivity Determination | At the frequency specified in the Surveillance Frequency Control Program. | 1, 2, 3, 4 |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | At the frequency specified in the Surveillance Frequency Control Program. | 1 |
| 3. Radiochemical for $\bar{E}$ Determination | At the frequency specified in the Surveillance Frequency Control Program**. | 1 |
| 4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135 | a. Once per 4 hours, whenever the specific activity exceeds 0.35 µCi/gram DOSE EQUIVALENT I-131 or $100/\bar{E}$ µCi/gram of gross radioactivity, and | 1#, 2#, 3#, 4#, 5# |
| | b. One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period. | 1, 2, 3 |

## TABLE 4.4-4 (Continued)

### TABLE NOTATIONS

**Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the Reactor Coolant System is restored within its limits.

## 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 limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued 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 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at the frequency specified in the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

## 3/4.4.9 PRESSURE/TEMPERATURE LIMITS

### LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the pressure limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS $T_{avg}$ and pressure at less than 200°F and 500 psig, respectively.

### SURVEILLANCE REQUIREMENTS

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

4.4.9.2.2 Deleted from Technical Specifications. Refer to the Technical Requirements Manual.

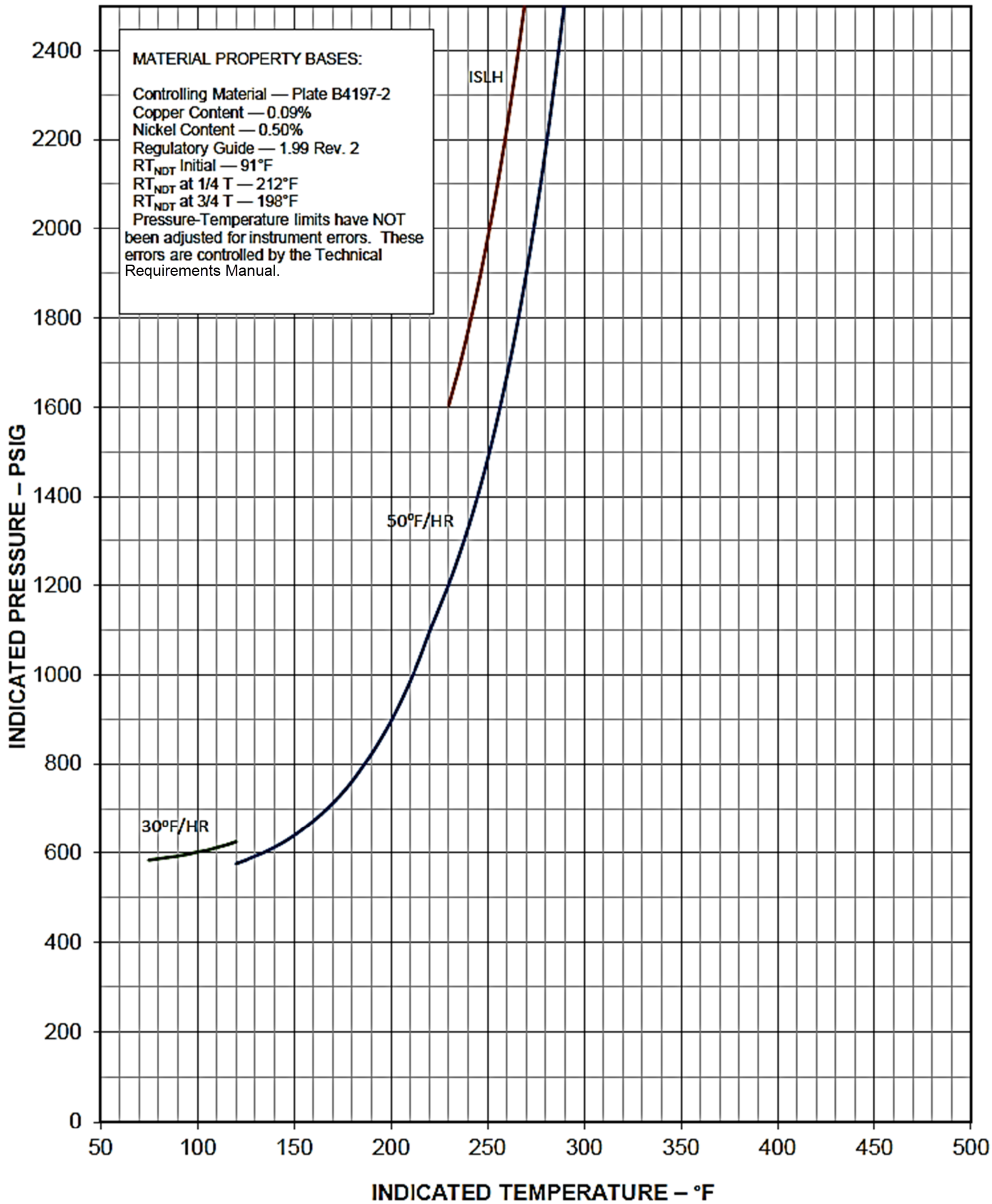


FIGURE 3.4-2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS – APPLICABLE UP TO 55 EFPY

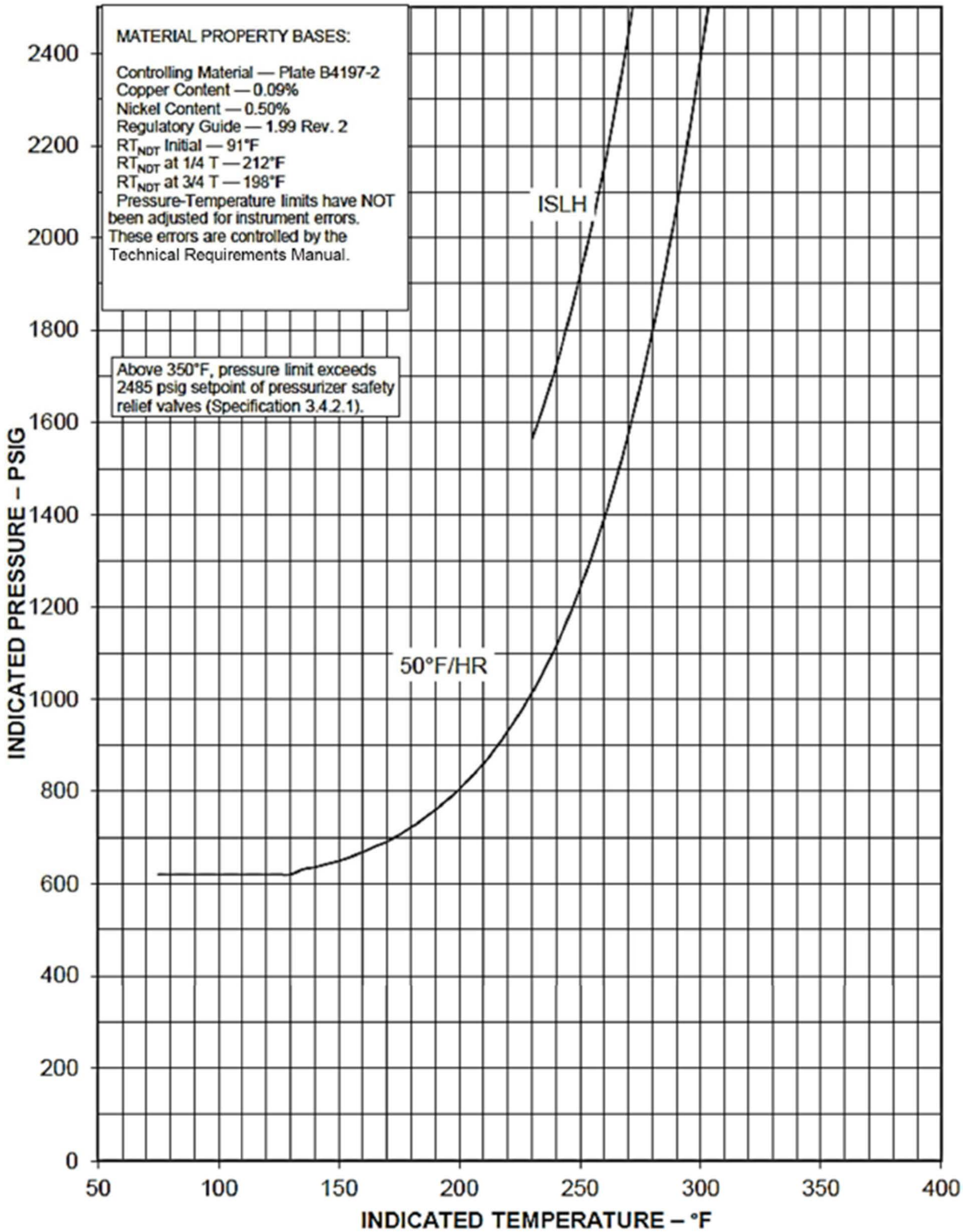


FIGURE 3.4-3
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS – APPLICABLE UP TO 55 EFPY

TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Requirements Manual.

## TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Specification Equipment List Program, plant procedure PLP-106.

## TABLE 4.4-6

## MAXIMUM COOLDOWN AND HEATUP RATES FOR MODES 4, 5, AND 6 (WITH REACTOR VESSEL HEAD ON)

### COOLDOWN RATES

| TEMPERATURE* | COOLDOWN IN ANY 1 HOUR PERIOD* |
|---|---|
| 350-120°F | 50°F |
| < 120°F | 30°F |

### HEATUP RATES

| TEMPERATURE* | HEATUP IN ANY 1 HOUR PERIOD* |
|---|---|
| <350°F | 50°F |

*Temperature used should be based on lowest RCS cold leg value except when no RCP is in operation; then use an operating RHR heat exchanger outlet temperature.

## PRESSURIZER

### LIMITING CONDITION FOR OPERATION

3.4.9.3 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,
b. A maximum cooldown of 200°F in any 1-hour period, and
c. A maximum spray water temperature differential of 625°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 structural integrity 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

4.4.9.3 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

## OVERPRESSURE PROTECTION SYSTEMS

### LIMITING CONDITION FOR OPERATION

3.4.9.4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.9 square inches, or

* b. Two power-operated relief valves (PORVs) with setpoints which do not exceed the limits established in Figure 3.4-4.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 325°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

----------------------------------------------------------NOTE----------------------------------------------------------

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

---------------------------------------------------------------------------------------------------------------------------

a. With one PORV inoperable in Mode 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.9 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.9 square inch vent within the next 8 hours.

c. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.9 square inch vent within 8 hours.

### SURVEILLANCE REQUIREMENTS

4.4.9.4.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to

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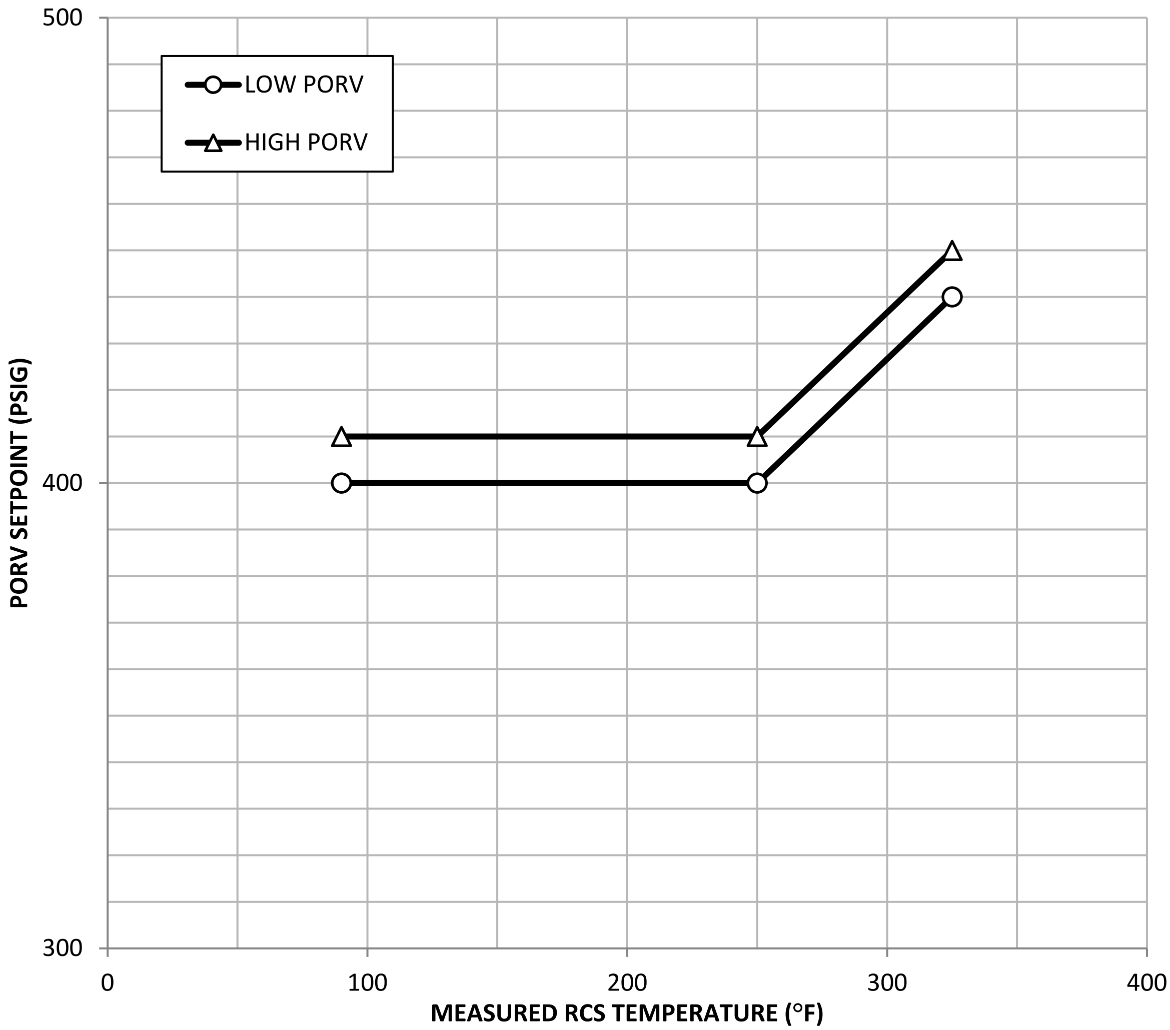
* Credit may only be taken for the setpoints when the RCS cold leg temperature ≥ 90°F.




| RCS TEMP (°F) | LOW PORV* (psig) | HIGH PORV* (psig) |
|---|---|---|
| 90 | 400 | 410 |
| 250 | 400 | 410 |
| 325 | 440 | 450 |

* VALUES BASED ON 55 EFPY REACTOR VESSEL DATA

INSTRUMENT ERRORS ARE CONTROLLED BY THE TECHNICAL REQUIREMENTS MANUAL.

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

entering a condition in which the PORV is required OPERABLE and at the frequency specified in the Surveillance Frequency Control Program when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at the frequency specified in the Surveillance Frequency Control Program; and

c. Verifying the PORV isolation valve is open at the frequency specified in the Surveillance Frequency Control Program when the PORV is being used for overpressure protection.

4.4.9.4.2 The RCS vent(s) shall be verified to be open at the frequency specified in the Surveillance Frequency Control Program* when the vent(s) is being used for overpressure protection.

* 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 at the frequency specified in the Surveillance Frequency Control Program.

3/4.4.10 STRUCTURAL INTEGRITY - DELETED

## 3/4.4.11 REACTOR COOLANT SYSTEM VENTS

### LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of at least one vent valve and one block valve, powered from emergency buses, shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space

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

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuators of all the vent valves in the inoperable vent path and both block valves; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable, due to causes other than the removal of power to both block valves pursuant to Action a, maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.4.11.1 (Section deleted)

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during venting.

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS

## 3/4.5.1 ACCUMULATORS

## COLD LEG INJECTION

### LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open with power supply circuit breaker open,

b. A contained borated water volume of between 66 and 96% indicated level,

c. A boron concentration within the limits specified in the COLR, and

d. A nitrogen cover-pressure of between 585 and 665 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 not within limits, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With one accumulator inoperable due to boron concentration not within limits, restore the boron concentration within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2. Verifying that each accumulator isolation valve is open.

*RCS pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At the frequency specified in the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of greater than or equal to 76 gallons, which is equivalent to an indicated level change of 9%, by verifying the boron concentration of the accumulator solution#; and

c. At the frequency specified in the Surveillance Frequency Control Program when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the respective isolation valve operator is open.

# This surveillance is not required when the volume increase makeup source is the Refueling Water Storage Tank (RWST) and the RWST has not been diluted since verifying that the RWST boron concentration is equal to or greater than the accumulator boron concentration limit.

## 3/4.5.2 ECCS SUBSYSTEMS - $T_{avg}$ GREATER THAN OR EQUAL TO 350°F

### LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

---------------------------------------------------------NOTE---------------------------------------------------------

*One ECCS subsystem train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

### SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the following valves are in the indicated positions with the control power disconnect switch in the "OFF" position, and the valve control switch in the "PULL TO LOCK" position:

| CP&L Valve No. | EBASCO Valve No. | Valve Function | Valve Position |
|---|---|---|---|
| 1SI-107 | 2SI-V500SA-1 | High Head Safety Injection to Reactor Coolant System Hot Legs | Closed |
| 1SI-86 | 2SI-V501SB-1 | High Head Safety Injection to Reactor Coolant System Hot Legs | Closed |
| 1SI-52 | 2SI-V502SA-1 | High Head Safety Injection to Reactor Coolant System Cold Legs | Closed |
| 1SI-340 | 2SI-V579SA-1 | Low Head Safety Injection to Reactor Coolant System Cold Legs | Open |
| 1SI-341 | 2SI-V578SB-1 | Low Head Safety Injection to Reactor Coolant System Cold Legs | Open |
| 1SI-359 | 2SI-V587SA-1 | Low Head Safety Injection to Reactor Coolant System Hot Legs | Closed |

b. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the ECCS locations susceptible to gas accumulation are sufficiently filled with water, and

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

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. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

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* Not required to be met for system vent flow paths opened under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

d. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying automatic interlock action of the RHR system from the Reactor Coolant System by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened.

2. 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 abnormal corrosion.

e. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on safety injection actuation test signal and on safety injection switchover to containment sump from an RWST Lo-Lo level test signal, and

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

a) Charging/safety injection pump,

b) RHR pump.

f. By verifying that each of the following pumps develops the required differential pressure when tested pursuant to the INSERVICE TESTING PROGRAM:

1. Charging/safety injection pump (Refer to Specification 4.1.2.4)

2. RHR pump ≥ 100 psid at a flow rate of at least 3663 gpm.

g. By verifying that the locking mechanism is in place and locked for the following High Head 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, and

2. At the frequency specified in the Surveillance Frequency Control Program.

| EBASCO Valve No. | CP&L Valve No. |
|---|---|
| 2SI-V440SA-1 | 1SI-5 |
| 2SI-V439SB-1 | 1SI-6 |
| 2SI-V438SA-1 | 1SI-7 |
| 2SI-V437SA-1 | 1SI-69 |
| 2SI-V436SB-1 | 1SI-70 |
| 2SI-V435SA-1 | 1SI-71 |
| 2SI-V434SA-1 | 1SI-101 |
| 2SI-V433SB-1 | 1SI-102 |
| 2SI-V432SA-1 | 1SI-103 |
| 2SI-V431SA-1 | 1SI-124 |
| 2SI-V430SB-1 | 1SI-125 |
| 2SI-V429SA-1 | 1SI-126 |

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 Charging/safety injection pump lines, with a single pump running:

   a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 348 gpm, and

   b) The total pump flow rate is less than or equal to 685 gpm.

2. For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3663 gpm.

## 3/4.5.3 ECCS SUBSYSTEMS - $T_{avg}$ LESS THAN 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 charging/safety injection pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

-----------------------------------------------------------NOTE-----------------------------------------------------------

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

-------------------------------------------------------------------------------------------------------------------------------

a. With no ECCS subsystem OPERABLE because of the inoperability of either the charging/safety injection 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 24 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR 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.

---

* A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F.

## SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 (Deleted).

## 3/4.5.4 REFUELING WATER STORAGE TANK

### LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 436,000 gallons, which is equivalent to 92% indicated level.

b. A boron concentration within the limits specified in the COLR,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 125°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour* or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the contained borated water volume in the tank, and
2. Verifying the boron concentration of the water.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 125°F.

* Except that while performing surveillance 4.4.6.2.2, the tank must be returned to OPERABLE status within 12 hours.

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 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

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that all penetrations*# not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. By performing required visual examinations and leakage rate testing, except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.

---

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

\# Valves CP-B3, CP-B7, and CM-B5 may be verified at the frequency specified in the Surveillance Frequency Control Program by manual remote keylock switch position.

## CONTAINMENT LEAKAGE

### LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be within the limits specified in the Containment Leakage Rate Testing Program.

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

ACTION:

With the containment leakage rate not within the limits specified in the Containment Leakage Rate Testing Program, restore the leakage rate to within the limits specified in the Containment Leakage Rate Testing Program prior to increasing the Reactor Coolant System temperature above 200°F.

### SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rate tests shall be performed in accordance with the Containment Leakage Rate Testing Program described in Technical Specification 6.8.4.k.

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LIMITING CONDITION FOR OPERATION

3.6.1.3 Two containment air locks shall be OPERABLE:

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. A separate ACTION is allowed for each air lock.
3. Enter 3.6.1.1 LCO for "Containment Integrity" when the air lock leakage results in exceeding the containment leakage rate, Specification 3.6.1.2.
4. Locking a Personnel Air Lock door shut consists of locking the associated manual pumping stations and deactivating the electronic mechanisms used to open a Personnel Air Lock door once the associated air lock door is shut. Locking an Emergency Air Lock door shut consists of locking the mechanical operator.

....................

a. One or more containment air locks with one containment air lock door inoperable:[#]
   1. Within one hour, verify the OPERABLE door is closed in the affected air lock, and
   2. Within 24 hours, lock the OPERABLE door closed in the affected air lock, and
   3. Once per 31 days, verify the OPERABLE door is locked closed in the affected air lock*, or
   4. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---

# 1. ACTIONS 3.6.1.3.a.1, 3.6.1.3.a.2, 3.6.1.3.a.3, and 3.6.1.3.a.4 are not applicable if both doors in the same air lock are inoperable and ACTION 3.6.1.3.c is entered.
  2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable.

* Air lock doors in high radiation areas may be verified closed by administrative means.

LIMITING CONDITION FOR OPERATION

b. One or more containment air locks with containment air lock interlock mechanism inoperable.##

   1. Within one hour, verify an OPERABLE door is closed in the affected air lock, and
   2. Within 24 hours, lock an OPERABLE door closed in the affected air lock, and
   3. Once per 31 days, verify the OPERABLE door is locked closed in the affected air lock*, or
   4. Otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. One or more containment air locks inoperable for reasons other than 3.6.1.3.a or 3.6.1.3.b.

   1. Immediately initiate action to evaluate containment leakage rate per LCO 3.6.1.2, and
   2. Within one hour, verify a door is closed in the affected air lock, and
   3. Within 24 hours or in accordance with the Risk-Informed Completion Time Program, restore air lock to OPERABLE status, or
   4. Otherwise be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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## 1. ACTIONS 3.6.1.3.b.1, 3.6.1.3.b.2, 3.6.1.3.b.3, and 3.6.1.3.b.4 are not applicable if both doors in the same air lock are inoperable and ACTION 3.6.1.3.c is entered.

2. Entry and exit of containment is permissible under the control of a dedicated individual.

* Air lock doors in high radiation areas may be verified closed by administrative means.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE by:

a. Performing required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program, as modified by the approved exemption###.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that only one door in the air lock can be opened at a time**.

---

### An inoperable air lock door does not invalidate the previous successful performance of the overall airlock leakage test.

** Only required to be performed upon entry or exit through the containment air lock. (If Surveillance Requirement 4.6.1.3.b has not been performed in the interval specified by the Surveillance Frequency Control Program, then perform Surveillance Requirement 4.6.1.3.b during the next containment entry through the associated air lock.)

## LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -1.0 inches water gauge and 1.6 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 at the frequency specified in 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 greater than 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 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at the frequency specified in the Surveillance Frequency Control Program, to be within the limit:

Location

a. Elevation 290 ft

b. Elevation 240 ft

c. Elevation 230 ft

## CONTAINMENT VESSEL STRUCTURAL INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel 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 vessel not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined, during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.1.c), by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation. Additional inspections shall be conducted in accordance with Subsections IWE and IWL of the ASME Boiler and Pressure Vessel Code, Section XI.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge makeup and exhaust isolation valve shall be OPERABLE and:

a. Each 42-inch containment preentry purge makeup and exhaust isolation valve shall be closed and sealed closed, and

b. The 8-inch containment purge makeup and exhaust isolation valve(s) may be open for safety-related reasons only.

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

ACTION:

a. With a 42-inch containment preentry purge makeup and/or exhaust isolation valve open or not sealed closed, close and/or seal close that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 8-inch containment purge makeup and/or exhaust isolation valve(s) inoperable for any reason other than leakage integrity, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge makeup and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specification 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT SYSTEMS
CONTAINMENT VENTILATION SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 42-inch containment preentry purge makeup and exhaust isolation valve shall be verified to be sealed closed and closed at the frequency specified in the Surveillance Frequency Control Program.

4.6.1.7.2 At the frequency specified in the Surveillance Frequency Control Program, the inboard and outboard valves in each makeup and exhaust penetration (2-42 inch valves and 2-8 inch valves) shall be demonstrated OPERABLE by verifying that the measured penetration leakage rate is less than 0.06 $L_a$ when pressurized to $P_a$.

CONTAINMENT SYSTEMS

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 containment sump.

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. Refer also to Specification 3.6.2.3 Action.

------------------------------------------------------------ NOTE ------------------------------------------------------------

**One Containment Spray System train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

## SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At the frequency specified in 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 an indicated recirculation flow of at least 1832 gpm, each pump develops a differential pressure of greater than or equal to 186 psi when tested pursuant to the INSERVICE TESTING PROGRAM;

c. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal, except for valves that are locked, sealed, or otherwise secured in the actuated position, and

   2. Verifying that each spray pump starts automatically on a containment spray actuation test signal.

   3. Verifying that, coincident with an indication of containment spray pump running, each automatic valve from the sump and RWST actuates to its appropriate position following an RWST Lo-Lo test signal, except for valves that are locked, sealed, or otherwise secured in the actuated position.

d. At the frequency specified in the Surveillance Frequency Control Program by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

e. At the frequency specified in the Surveillance Frequency Control Program by verifying that containment spray locations susceptible to gas accumulation are sufficiently filled with water.

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* Not required to be met for system vent flow paths opened under administrative control.

## LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A Spray Additive Tank containing a volume of between 3268 and 3768 gallons of between 27 and 29 weight % of NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

## SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At the frequency specified in 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. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

c. At the frequency specified in the Surveillance Frequency Control Program by verifying that each automatic valve in the flow path actuates to its correct position on a containment spray or containment isolation phase A test signal as applicable; except for valves that are locked, sealed, or otherwise secured in the actuated position, and

d. At the frequency specified in the Surveillance Frequency Control Program by verifying each eductor flow rate is between 17.2 and 22.2 gpm, using the RWST as the test source containing at least 436,000 gallons of water.

## LIMITING CONDITION FOR OPERATION

3.6.2.3 Four containment fan coolers (AH-1, AH-2, AH-3, and AH-4) shall be OPERABLE with one of two fans in each cooler capable of operation at low speed. Train SA consists of AH-2 and AH-3. Train SB consists of AH-1 and AH- 4.

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

ACTION:

a. With one train of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore the inoperable train of fan coolers to OPERABLE status within 7 days or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both trains of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore at least one train of fan coolers to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required trains of fan coolers to OPERABLE status within 7 days of initial loss or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one train of the above required containment fan coolers inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable train of containment fan coolers to OPERABLE status within 7 days of initial loss or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---------------------------------------------------------- NOTE ----------------------------------------------------------

*One train of containment fan coolers and one Containment Spray System train are allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

## SURVEILLANCE REQUIREMENTS

4.6.2.3 Each train of containment fan coolers shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Starting each fan train from the control room, and verifying that each fan train operates for at least 15 minutes, and

   2. Verifying a cooling water flow rate, after correction to design basis service water conditions, of greater than or equal to 1300 gpm to each cooler.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that each fan train starts automatically on a safety injection test signal.

## 3/4.6.3 CONTAINMENT ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve specified in the Technical Requirements Manual shall be OPERABLE with isolation times less than or equal to required isolation times.

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

ACTION:

With one or more of the containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours or in accordance with the Risk-Informed Completion Time Program, or

b. Isolate each affected penetration within 4 hours or in accordance with the Risk-Informed Completion Time Program by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours or in accordance with the Risk-Informed Completion Time Program 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.

### SURVEILLANCE REQUIREMENTS

4.6.3.1 Each isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each normal, preentry purge makeup and exhaust, and containment vacuum relief valve actuates to its isolation position, and

d. Verifying that, on a Safety Injection "S" test signal, each containment isolation valve receiving an "S" signal actuates to its isolation position, and

e. Verifying that, on a Main Steam Isolation test signal, each main steam isolation valve actuates to its isolation position, and

f. Verifying that, on a Main Feedwater Isolation test signal, each feedwater isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit specified in the Technical Requirements Manual when tested pursuant to the INSERVICE TESTING PROGRAM.

TABLE 3.6-1 CONTAINMENT ISOLATION VALVES

This table is deleted from Technical Specifications.

The information in this table is controlled by the Technical Requirements Manual.

PAGES 3/4 6-17 THROUGH 3/4 6-29 HAVE BEEN DELETED.

## 3/4.6.4 COMBUSTIBLE GAS CONTROL

### HYDROGEN MONITORS

### LIMITING CONDITION FOR OPERATION

3.6.4.1 Deleted.

## ELECTRIC HYDROGEN RECOMBINERS

### LIMITING CONDITION FOR OPERATION

3.6.4.2 Deleted.

## 3/4.6.5 VACUUM RELIEF SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.5 The containment vacuum relief system shall be OPERABLE with an Actuation Setpoint of equal to or less negative than -2.5 inches water gauge differential pressure (containment pressure less atmospheric pressure)

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

ACTION:

With one containment vacuum relief system inoperable, restore the system 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

4.6.5 No additional requirements other than those required by the INSERVICE TESTING PROGRAM.

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 associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more main steam line Code safety valves inoperable, operation may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by the INSERVICE TESTING PROGRAM.

## TABLE 3.7-1

## MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

| MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR | MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER) |
|---|---|
| 1 | 49 |
| 2 | 32 |
| 3 | 15 |

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

| VALVE NUMBER | | | LIFT SETTING (± 3%)* | ORIFICE SIZE (IN.$^2$) |
|---|---|---|---|---|
| STEAM GENERATOR | | | | |
| A | B | C | | |
| 1MS-43 | 1MS-44 | 1MS-45 | 1170 psig | 16.0 |
| 1MS-46 | 1MS-47 | 1MS-48 | 1185 psig | 16.0 |
| 1MS-49 | 1MS-50 | 1MS-51 | 1200 psig | 16.0 |
| 1MS-52 | 1MS-53 | 1MS-54 | 1215 psig | 16.0 |
| 1MS-55 | 1MS-56 | 1MS-57 | 1230 psig | 16.0 |

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

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

----------------------------------------------------------NOTE----------------------------------------------------------

LCO 3.0.4.b is not applicable.

-------------------------------------------------------------------------------------------------------------------------

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program 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. (NOTE: LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. Following restoration of one AFW train, all applicable LCOs apply based on the time the LCOs initially occurred.)

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Demonstrating that each motor-driven pump satisfies performance requirements by either:

a) Verifying each pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1514 psid at a recirculation flow of greater than or equal to 50 gpm (25 KPPH), or

b) Verifying each pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1259 psid at a flow rate of greater than or equal to 430 gpm (215 KPPH).

SURVEILLANCE REQUIREMENTS (CONTINUED)

2. Demonstrating that the steam turbine - driven pump satisfies performance requirements by either:

*****************************************************************

NOTE: The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

*****************************************************************

a) Verifying the pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1167 psid at a recirculation flow of greater than or equal to 81 gpm (40.5 KPPH) when the secondary steam supply pressure is greater than 210 psig, or

b) Verifying the pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1400 psid at a flow rate of greater than or equal to 430 gpm (215 KPPH) when the secondary steam supply pressure is greater than 280 psig.

b. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying by flow or position check 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; and

2. Verifying that the isolation valves in the suction line from the CST are locked open.

c. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that each motor-driven auxiliary feedwater pump starts automatically, as designed, upon receipt of a test signal and that the respective pressure control valve for each motor-driven pump and each flow control valve with an auto-open feature respond as required;

2. Verifying that the turbine-driven auxiliary feedwater pump starts automatically, as designed, upon receipt of a test signal. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3; and

3. Verifying that the motor-operated auxiliary feedwater isolation valves and flow control valves close as required upon receipt of an appropriate test signal for steamline differential pressure high coincident with main steam isolation.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

## LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained water volume of at least 270,000 gallons of water, which is equivalent to 62% indicated level.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the Emergency Service Water System as a backup supply to the auxiliary feedwater pumps and restore the CST to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

## SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The CST shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The Emergency Service Water System shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying that each valve, required to permit the Emergency Service Water System to supply water to the auxiliary feedwater pumps, is open whenever the Emergency Service Water System is the supply source for the auxiliary feedwater pumps.

SPECIFIC ACTIVITY

## LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microCurie/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microCurie/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 4.7-1.

## TABLE 4.7-1

## SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY |
|---|---|
| 1. Gross Radioactivity Determination*<br>or<br>Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | At the frequency specified in the Surveillance Frequency Control Program. |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | a. Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines.<br>b. At the frequency specified in the Surveillance Frequency Control Program, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines. |

*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than 15 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

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

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours or in accordance with the Risk-Informed Completion Time Program; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2, 3, and 4:

With one MSIV inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV 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 for entry into MODES 3 or 4.

## 3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

### LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 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 less than or equal 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 The pressure in each side of the steam generator shall be determined to be less than 200 psig at the frequency specified in the Surveillance Frequency Control Program when the temperature of either the reactor or secondary coolant is less than 70°F.

## 3/4.7.3 COMPONENT COOLING WATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.3 At least two component cooling water (CCW) pumps*, heat exchangers and essential flow paths shall be OPERABLE.

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

ACTION:

With only one component cooling water flow path OPERABLE, restore at least two flow paths to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program 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 At least two component cooling water flow paths shall be demonstrated OPERABLE:

a. At the frequency specified in 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; and

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

1. Each automatic valve servicing safety-related equipment or isolating non-safety-related components actuates to its correct position on a Safety Injection test signal, except for valves that are locked, sealed, or otherwise secured in the actuated position, and
2. Each Component Cooling Water System pump required to be OPERABLE starts automatically on a Safety Injection test signal.
3. Each automatic valve serving the gross failed fuel detector and sample system heat exchangers actuates to its correct position on a Low Surge Tank Level test signal, except for valves that are locked, sealed, or otherwise secured in the actuated position.

* The breaker for CCW pump 1C-SAB shall not be racked into either power source (SA or SB) unless the breaker from the applicable CCW pump (1A-SA or 1B-SB) is racked out.

## 3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent emergency service water loops shall be OPERABLE.

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

ACTION:

With only one emergency service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

------------------------------------------------------------ NOTE ------------------------------------------------------------

*The ‘B’ Train emergency service water loop is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

### SURVEILLANCE REQUIREMENTS

4.7.4 At least two emergency service water loops shall be demonstrated OPERABLE:

a. At the frequency specified in 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; and

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

   1. Each automatic valve servicing safety-related equipment or isolating non-safety portions of the system actuates to its correct position on a Safety Injection test signal, except for valves that are locked, sealed, or otherwise secured in the actuated position, and

   2. Each emergency service water pump and each emergency service water booster pump starts automatically on a Safety Injection test signal.

## 3/4.7.5 ULTIMATE HEAT SINK

### LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum auxiliary reservoir water level at or above elevation 250 feet Mean Sea Level, USGS datum, and a minimum main reservoir water level at or above 206 feet Mean Sea Level, USGS datum, and

b. A water temperature as measured at the respective intake structure of less than or equal to 94°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying the water temperature and water level to be within their limits.

## LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Room Emergency Filtration System (CREFS) trains shall be OPERABLE.*

APPLICABILITY:
a. MODES 1, 2, 3, and 4
b. MODES 5 and 6
c. During movement of irradiated fuel assemblies and movement of loads over spent fuel pools

ACTION:

a. MODES 1, 2, 3 and 4:

-------------------------------------------NOTE----------------------------------------
In addition to the Actions below, perform Action c. if applicable.
-----------------------------------------------------------------------------------------

1. With one CREFS train inoperable for reasons other than an inoperable Control Room Envelope (CRE) boundary, restore the inoperable CREFS train 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.

2. With one or more CREFS trains inoperable due to inoperable CRE boundary:

   a. Initiate action to implement mitigating actions immediately or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours;

   b. Within 24 hours, verify mitigating actions ensure CRE occupant radiological exposures will not exceed limits and that CRE occupants are protected from hazardous chemicals and smoke or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours;

   c. Restore CRE boundary to OPERABLE within 90 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* The control room envelope (CRE) boundary may be opened intermittently under administrative controls.

LIMITING CONDITION FOR OPERATION (Continued)

b. MODES 5 and 6

-------------------------------------------NOTE----------------------------------------

In addition to the Actions below, perform Action c. if applicable.

------------------------------------------------------------------------------------------

1. With one CREFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable CREFS train to OPERABLE status within 7 days or immediately initiate and maintain operation of the remaining OPERABLE CREFS train in the recirculation mode.
2. With both CREFS trains inoperable for reasons other than an inoperable CRE boundary or with the OPERABLE CREFS train required to be in the recirculation mode by ACTION b.1., not capable of being powered by an OPERABLE emergency power source, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel.
3. With one or more CREFS trains inoperable due to inoperable CRE boundary, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies.

## LIMITING CONDITION FOR OPERATION (Continued)

c. During movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

   1. With one CREFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable CREFS train to OPERABLE status within 7 days or immediately initiate and maintain operation of the remaining OPERABLE CREFS train in the recirculation mode; or immediately suspend movement of irradiated fuel.

   2. With both CREFS trains inoperable for reasons other than an inoperable CRE boundary, or with the OPERABLE CREFS train required to be in the recirculation mode by Action c.1., not capable of being powered by an OPERABLE emergency power source, immediately suspend all operations involving movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

   3. With one or more CREFS trains inoperable due to inoperable CRE boundary, immediately suspend movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

## SURVEILLANCE REQUIREMENTS

4.7.6 Each CREFS train shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 continuous minutes;

b. At the frequency specified in the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980; and

SURVEILLANCE REQUIREMENTS (CONTINUED)

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤0.5% when tested at a temperature of 30°C and at a relative humidity of 95% in accordance with ASTM D3803 -1989.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤0.5% when tested at a temperature of 30°C and at a relative humidity of 95% in accordance with ASTM D3803-1989.

d. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.1 inches water gauge while operating the system at a flow rate of 4000 cfm ± 10%;

2. Verifying that, on either a Safety Injection or a High Radiation test signal, the system automatically switches into an isolation with recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position;

3. Deleted.

4. Deleted.

5. Deleted.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 4000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

g. Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

## 3/4.7.7 REACTOR AUXILIARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.7 Two independent RAB Emergency Exhaust Systems shall be OPERABLE.*

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

ACTION:

a. With one RAB Emergency Exhaust System inoperable, restore the inoperable system 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.

b. With two RAB Emergency Exhaust Systems inoperable due to an inoperable RAB Emergency Exhaust System boundary, restore the RAB Emergency Exhaust System boundary to OPERABLE status within 24 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.7 Each RAB Emergency Exhaust System shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 continuous minutes;

b. At the frequency specified in the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6800 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980;

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodine penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 95% in accordance with ASTM D3803-1989.

* The RAB Emergency Exhaust Systems boundary may be opened intermittently under administrative controls.

SURVEILLANCE REQUIREMENTS (CONTINUED)

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 95% in accordance with ASTM D3803-1989.

d. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is less than 4.1 inches water gauge while operating the unit at a flow rate of 6800 cfm ± 10%,

   2. Verifying that the system starts on a Safety Injection test signal,

   3. Verifying that the system maintains the areas served by the exhaust system at a negative pressure of greater than or equal to 1/8 inch water gauge relative to the outside atmosphere,

   4. Verifying that the filter cooling bypass valve is locked in the balanced position, and

   5. Deleted.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the unit at a flow rate of 6800 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6800 cfm ± 10%.

## 3/4.7.8 SNUBBERS

### LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

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 on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per augmented inservice inspection program on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

### SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the augmented inservice inspection program specified in the Technical Requirements Manual.

PAGES 3/4 7-20 THROUGH 3/4 7-23 HAVE BEEN DELETED.

FIGURE 4.7-1 SAMPLE PLAN (2) FOR SNUBBER FUNCTIONAL TEST

This figure is deleted from Technical Specifications and is controlled by the Technical Requirements Manual.

PLANT SYSTEMS

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source (excluding startup sources and fission detectors previously subjected to core flux) containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 10 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

    1. Decontaminate and repair the sealed source, or

    2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 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 microCurie per test sample.

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

    1. With a half-life greater than 30 days (excluding Hydrogen 3), and

    2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

3/4.7.10 FIRE SUPPRESSION - DELETED
3/4.7.11 FIRE RATED ASSEMBLIES - DELETED
TABLES 3.7-3, 3.7-4, 3.7-5 - DELETED

## 3/4.7.12 AREA TEMPERATURE MONITORING - DELETED

TABLE 3.7-6

AREA TEMPERATURE MONITORING - DELETED

## 3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.13 At least two independent Essential Services Chilled Water System loops shall be OPERABLE.

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

ACTION:

With only one Essential Services Chilled Water System loop OPERABLE, restore at least two loops to OPERABLE status within 7 days* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.13 The Essential Services Chilled Water System shall be demonstrated OPERABLE by:

a. Performance of surveillances as required by the INSERVICE TESTING PROGRAM, and

b. At the frequency specified in the Surveillance Frequency Control Program by demonstrating that:

1. Non-essential portions of the system are automatically isolated upon receipt of a Safety Injection actuation signal, except for valves that are locked, sealed, or otherwise secured in the actuated position, and
2. The system starts automatically on a Safety Injection actuation signal.

*Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

## 3/4.7.14 FUEL STORAGE POOL BORON CONCENTRATION

### LIMITING CONDITION FOR OPERATION

3.7.14 The boron concentration of spent fuel pools shall be ≥ 2000 ppm.

APPLICABILITY: At ALL TIMES for pools that contain nuclear fuel.

ACTION:

a. With the spent fuel pool boron concentration not within the limits, immediately suspend movement of fuel assemblies.

b. Immediately initiate action to restore pool boron concentration within the limit.

### SURVEILLANCE REQUIREMENTS

4.7.14 At the frequency specified in the Surveillance Frequency Control Program verify spent fuel pool boron concentration is within the limit by:

a. Sampling the water volume connected to or in applicable pools.

b. In addition to 4.7.14.a, sampling an individual pool containing nuclear fuel if the pool is isolated from other pools.

## 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 circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. A separate day tank containing a minimum of 1457 gallons of fuel,

2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and

3. A separate fuel oil transfer pump.

c. Automatic Load Sequencers for Train A and Train B.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

------------------------------------------------------------NOTE------------------------------------------------------------

LCO 3.0.4.b is not applicable to diesel generators.

--------------------------------------------------------------------------------------------------------------------------------

a. With one offsite circuit of 3.8.1.1.a inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

2. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

3. Verify required feature(s) powered from the OPERABLE offsite A.C. source are OPERABLE. If required feature(s) powered from the OPERABLE offsite circuit are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 24 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

b. With one diesel generator of 3.8.1.1.b inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

*2. Within 24 hours, determine the OPERABLE diesel generator is not inoperable due to a common cause failure or perform Surveillance Requirement 4.8.1.1.2.a.4#; and

---

* This ACTION is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

# Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

3. Restore the diesel generator to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. Verify required feature(s) powered from the OPERABLE diesel generator are OPERABLE. If required feature(s) powered from the OPERABLE diesel generator are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 4 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

c. With one offsite circuit and one diesel generator of 3.8.1.1 inoperable:

NOTE: Enter applicable Condition(s) and Required Action(s) of LCO 3/4.8.3, ONSITE POWER DISTRIBUTION - OPERATING, when this condition is entered with no A.C. power to one train.

1. Restore one of the inoperable A.C. sources to OPERABLE status within 12 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. Following restoration of one A.C. source (offsite circuit or diesel generator), restore the remaining inoperable A.C. source to OPERABLE status pursuant to requirements of either ACTION a or b, based on the time of initial loss of the remaining A.C. source.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

d. With two of the required offsite A.C. sources inoperable:

1. Restore one offsite circuit to OPERABLE status within 24 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

2. Verify required feature(s) are OPERABLE. If required feature(s) are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 12 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) inoperable.

3. Following restoration of one offsite A.C. source, restore the remaining offsite A.C. source in accordance with the provisions of ACTION a with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. source.

e. With two of the required diesel generators inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

#2. Restore one of the 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.

3. Following restoration of one diesel generator, restore the remaining diesel generator in accordance with the provisions of ACTION b with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable diesel generator.

f. With three or more of the required A.C. sources inoperable:

1. Immediately enter Technical Specification 3.0.3.

2. Following restoration of one or more A.C. sources, restore the remaining inoperable A.C. sources in accordance with the provisions of ACTION a, b, c, d and/or e as applicable with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. sources.

g. Deleted.

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#Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

h. With one automatic load sequencer inoperable:

1. Restore the automatic load sequencer to OPERABLE status within 24 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and power availability, and

b. Demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by manually transferring the onsite Class 1E power supply from the unit auxiliary transformer to the startup auxiliary transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the fuel level in the day tank,
2. Verifying the fuel level in the main fuel oil storage tank,
3. Verifying the fuel oil transfer pump can be started and transfers fuel from the storage system to the day tank,
4. Verifying the diesel generator can start** and accelerate## to synchronous speed (450 rpm) with generator steady-state voltage and frequency 6900 ± 276 volts and 60 ± 0.48 Hz,
5. Verifying the diesel generator is synchronized, gradually loaded** to an indicated 6200-6400 kW*** and operates for at least 60 minutes,
6. Verifying the pressure in at least one air start receiver to be greater than or equal to 190 psig, and
7. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

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** This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable, regarding loading recommendations.

*** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

## The voltage and frequency conditions shall be met within 10 seconds or gradual acceleration to no-load conditions per vendor recommendations will be an acceptable alternative.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.8.1.1.2 (Continued)

b. Check for and remove accumulated water:

1. From the day tank, at the frequency specified in the Surveillance Frequency Control Program and after each operation of the diesel where the period of operation was greater than 1 hour, and

2. From the main fuel oil storage tank, at the frequency specified in the Surveillance Frequency Control Program.

c. By verifying fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program, at frequencies in accordance with the Diesel Fuel Oil Testing Program.

d. DELETED.

e. At the frequency specified in the Surveillance Frequency Control Program, the diesel generators shall be started** and accelerated to at least 450 rpm in less than or equal to 10 seconds. The generator steady-state voltage and frequency shall be 6900 ± 276 volts and 60 ± 0.48 Hz in less than or equal to 10 seconds after the start signal.

** This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.8.1.1.2 (Continued)

The generator shall be manually synchronized to its appropriate emergency bus, loaded to an indicated 6200-6400***kW, and operate for at least 60 minutes. The diesel generator shall be started for this test by using one of the following signals on a rotating basis:

1. Simulated loss of offsite power by itself, and
2. A Safety Injection test signal by itself.

This test, if it is performed so that it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4, may also serve to concurrently meet those requirements as well.

f. At the frequency specified in the Surveillance Frequency Control Program by:

1. DELETED
2. [1]Verifying that, on rejection of a load of greater than or equal to 1078 kW, the voltage and frequency are maintained with 6900 ± 690 volts and 60 ± 6.75 Hz, with voltage stabilizing to 6900 ± 276 volts and frequency stabilizing to 60 ± 0.48 Hz within 10 seconds without any safety-related load tripping out or operating in a degraded condition.
3. [2]Verifying that the load sequencing timer is OPERABLE with the interval between each load block within 10% of its design interval.
4. [3]Verifying on an actual or simulated loss of offsite power signal by itself:

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*** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

1 This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

2 This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

3 This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 (Continued)

a) De-energization of the emergency buses and load shedding from the emergency buses.

b) The diesel starts** on the auto-start signal, energizing the emergency buses with permanently connected loads in less than or equal to 10 seconds, energizing the auto-connected shutdown loads through the load sequencer, and operating 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 shall be maintained at 6900 ± 276 volts and 60 ± 0.48 Hz.

5. [1]Verifying that on an actual or simulated safety injection signal (without loss of power) the diesel generator starts** on the auto-start signal and operates on standby for greater than or equal to 5 minutes.

6. [3]Verifying on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated safety injection signal:

a) De-energization of the emergency buses and load shedding from the emergency buses.

b) The diesel starts** on the auto-start signal, energizing the emergency buses with permanently connected loads in less than or equal to 10 seconds, energizing the auto-connected emergency (accident) loads through the sequencing timers, and operating for greater than or equal to 5 minutes and maintaining the steady-state voltage and frequency at 6900 ± 276 volts and 60 ± 0.48 Hz.

c) DELETED

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** This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

[1] This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

[3] This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 (Continued)

7. Verifying the diesel generator operates** for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to 6800-7000 kW*** and, during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 6200-6400 kW***.
8. DELETED
9. [2]Verifying the diesel generator's capability to:
   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
   b) Transfer its loads to the offsite power source, and
   c) Proceed through its shutdown sequence.
10. DELETED
11. [1]Verifying the generator capability to reject a load of between 6200 and 6400 kW without tripping. The generator voltage shall not exceed 8280 volts during and following the load rejection.
12. [3]Verifying that, with the diesel generator operating in a test mode and connected to its bus, an actual or 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.

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** This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

*** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

[1] This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

[2] This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

[3] This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.8.1.1.2 (Continued)

13. [1]Verifying that all diesel generator trips, except engine overspeed, loss of generator potential transformer circuits, generator differential, and emergency bus differential are automatically bypassed on a simulated or actual loss of offsite power signal in conjunction with a safety injection signal.

14. Verifying that within 5 minutes of shutting down the EDG, after the EDG has operated for at least 2 hours at an indicated load of 6200-6400 kW, the EDG starts and accelerates to a steady-state voltage and frequency of 6900 ± 276 volts and 60 ± 0.48 Hz in 10 seconds or less.

g. At the frequency specified in the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting** both diesel generators simultaneously from standby condition and verifying that both diesel generators accelerate to at least 450 rpm in less than or equal to 10 seconds.

h. At the frequency specified in the Surveillance Frequency Control Program by:

1) DELETED.

2) Performing a pressure test, of those isolable portions of the diesel fuel oil piping system designed to Section III, subsection ND of the ASME Code, at a test pressure equal to 110% of the system design pressure.

** This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

1 This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

TABLE 4.8-1 HAS BEEN DELETED

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C electrical power sources shall be OPERABLE

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day tank containing a minimum volume of 1457 gallons of fuel,

2. A separate main fuel oil storage tank containing a minimum volume of 100,000 gallons of fuel, and

3. A fuel oil transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over irradiated fuel and within 8 hours, depressurize and vent the Reactor Coolant System through a vent of greater than or equal to 2.9 square inches. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1 and 4.8.1.1.2.

## LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Emergency Battery Bank 1A-SA and either full capacity charger, 1A-SA or 1B-SA, and,

b. 125-volt Emergency Battery Bank 1B-SB and either full capacity charger, 1A-SB or 1B-SB.

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

ACTION:

With one of the required D.C. electrical sources inoperable, restore the inoperable D.C. electrical source to OPERABLE status within 2 hours or in accordance with the Risk-Informed Completion Time Program 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 Each 125-volt Emergency Battery and charger shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

b. At the frequency specified in the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than $150 \times 10^{-6}$ ohm, and

3. The average electrolyte temperature of 10 connected cells is above 70° F.

SURVEILLANCE REQUIREMENTS (CONTINUED)

c. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

   1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

   2. The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

   3. The resistance of each cell-to-cell and terminal connection is less than or equal to $150 \times 10^{-6}$ ohm, and

   4. The battery charger will supply at least 150 amperes at greater than or equal to 125 volts for at least 4 hours.

d. #At the frequency specified in the Surveillance Frequency Control Program by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. #At the frequency specified in the Surveillance Frequency Control Program by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. At the frequency specified in the Surveillance Frequency Control Program, this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1.d; and

f. #At least once per 18 months by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

# This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

## TABLE 4.8-2

## BATTERY SURVEILLANCE REQUIREMENTS

| | CATEGORY A[1] | CATEGORY B[2] | |
|---|---|---|---|
| PARAMETER | LIMITS FOR EACH DESIGNATED PILOT CELL | LIMITS FOR EACH CONNECTED CELL | ALLOWABLE[3] VALUE FOR EACH CONNECTED CELL |
| Electrolyte Level | >Minimum level indication mark, and < ¼" above maximum level indication mark | >Minimum level indication mark, and < ¼" above maximum level indication mark | Above top of plates, and not overflowing |
| Float Voltage | $\geq$ 2.13 volts | $\geq$ 2.13 volts [6] | > 2.07 volts |
| Specific Gravity[4] | $\geq$ 1.200[5] | $\geq$ 1.195 | Not more than 0.020 below the average of all connected cells |
| | | Average of all connected cells > 1.205 | Average of all connected cells $\geq$ 1.195[5] |

### TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

## LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt Emergency Battery (either 1A-SA or 1B-SB) and at least one associated full-capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required Emergency Battery or full-capacity charger inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required Emergency Battery and full-capacity charger to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a vent of $\geq$ 2.9 square inches.

## SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt Emergency Battery and full-capacity charger shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

## 3/4.8.3 ONSITE POWER DISTRIBUTION

### OPERATING

### LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical buses shall be energized in the specified manner with tie breakers open between redundant buses within the unit:

a. Division A ESF A.C. Buses consisting of:

1. 6900-volt Bus 1A-SA.
2. 480-volt Bus 1A2-SA.
3. 480-volt Bus 1A3-SA.

b. Division B ESF A.C. Buses consisting of:

1. 6900-volt Bus 1B-SB.
2. 480-volt Bus 1B2-SB.
3. 480-volt Bus 1B3-SB.

c. 118-volt A.C. Vital Bus 1DP-1A-SI energized from its associated inverter connected to 125-volt D.C. Bus DP-1A-SA*,

d. 118-volt A.C. Vital Bus 1DP-1A-SIII energized from its associated inverter connected to 125-volt D.C. Bus DP-1A-SA*,

e. 118-volt A.C. Vital Bus 1DP-1B-SII energized from its associated inverter connected to 125-volt D.C. Bus DP-1B-SB*,

f. 118-volt A.C. Vital Bus 1DP-1B-SIV energized from its associated inverter connected to 125-volt D.C. Bus DP-1B-SB*,

g. 125-volt D.C. Bus DP-1A-SA energized from Emergency Battery 1A-SA and charger 1A-SA or 1B-SA, and

h. 125-volt D.C. Bus DP-1B-SB energized from Emergency Battery 1B-SB and charger 1B-SB or 1A-SB

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

*Two inverters may be disconnected from their 125-volt D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated Emergency Battery provided: (1) their vital buses are energized and (2) the vital buses associated with the other Emergency Battery are energized from their associated inverters and connected to their associated 125-volt D.C. bus.

LIMITING CONDITION FOR OPERATION

ACTION:

a. With one of the required divisions of A.C. ESF buses not fully energized, reenergize the division within 8 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 118-volt A.C. vital bus not energized from its associated inverter, reenergize the 118-volt A.C. vital bus within 2 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one 118-volt A.C. vital bus not energized from its associated inverter connected to its associated D.C. bus, re-energize the 118-volt A.C. vital bus through its associated inverter connected to its associated D.C. bus within 24 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With either 125-volt D.C. bus 1A-SA or 1B-SB not energized from its associated Emergency Battery, reenergize the D.C. bus from its associated Emergency Battery within 2 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified buses shall be determined energized in the required manner at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the buses.

ELECTRICAL POWER SYSTEMS
ONSITE POWER DISTRIBUTION
SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, one of the following divisions of electrical buses shall be energized in the specified manner:

a. Division A, consisting of:
   1. 6900-volt Bus 1A-SA and
   2. 480-volt Buses 1A2-SA and 1A3-SA, and
   3. 118-volt A.C. Vital Buses 1DP-1A-SI and 1DP-1A-SIII energized from their associated inverter connected to 125-volt D.C. Bus DP-1A-SA, and
   4. 125-volt D.C. Bus DP-1A-SA energized from Emergency Battery 1A-SA and chargers 1A-SA or 1B-SA, or

b. Division B, consisting of:
   1. 6900-volt Bus 1B-SB and
   2. 480-volt Buses 1B2-SB and 1B3-SB, and
   3. 118-volt AC Vital Buses 1DP-1B-SII and 1DP-1B-SIV energized from their associated inverter connected to 125-volt D.C. Bus DP-1B-SB, and
   4. 125-volt D.C. Bus DP-1B-SB energized from Emergency Battery 1B-SB and chargers 1B-SB or 1A-SB.

APPLICABILITY MODES 5 and 6.

ACTION:

With any of the above required electrical buses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to energize the required electrical buses in the specified manner as soon as possible; and within 8 hours, depressurize and vent the RCS through a vent of ≥ 2.9 square inches.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified buses shall be determined energized in the required manner at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the buses.

Specifications 3/4.8.4.1 and 3/4.8.4.2 have been deleted from Technical Specifications and relocated to the Technical Requirements Manual.

PAGES 3/4 8-20 THROUGH 3/4 8-43 HAVE BEEN DELETED.

Pages 3/4 8-20, 3/4 8-21, 3/4 8-39, and 3/4 8-40 by Amendment No. 182.

Pages 3/4 8-22 through 3/4 8-38B and 3/4 8-41 through 3/4 8-43 by Amendment No. 13.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.a The boron concentration of all filled portions of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained uniform and within the limit specified in the COLR.

3.9.1.b The valves listed in Table 3.9-1 shall be in their positions required by Table 3.9-1.

APPLICABILITY: MODE 6.

ACTION:

a. With the requirements of Specification 3.9.1.a not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes, and initiate actions to restore boron concentration to within limits.

b. With the requirements of Specification 3.9.1.b not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes, and initiate action to return the valve(s) to the position required by Table 3.9-1.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be determined by chemical analysis to be within the limits of the COLR at the frequency specified in the Surveillance Frequency Control Program.

4.9.1.2 At the frequency specified in the Surveillance Frequency Control Program, verify that the valves listed in Table 3.9-1 are in their positions required by Table 3.9-1.

## TABLE 3.9-1

## ADMINISTRATIVE CONTROLS TO PREVENT DILUTION DURING REFUELING

| CP&L VALVE NO. (Ebasco Valve No.) | DESCRIPTION | REQUIRED POSITION |
|---|---|---|
| 1CS-149 (CS-D121SN) | Reactor Makeup Water to CVCS Makeup Control System | Lock closed; may be opened to permit makeup to Refueling Water Storage Tank provided valves 1CS-156 and 1CS-155 are maintained closed with their main control board control switches in "shut" position, and manual valves 1CS-274, 1CS-265 and 1CS-287 are locked closed. |
| 1CS-510 (CS-D631SN) | Boric Acid Batch Tank Outlet | Locked closed; may be opened provided the boron concentration of the boric acid batch tank ≥ the greater of 2000 ppm or the boron concentration required to maintain $K_{eff}$ less than or equal to 0.95, as specified in the COLR and valve 1CS-503 is closed. |
| 1CS-503 (CS-D251) | Reactor Makeup Water to Boric Acid Batch Tank | Lock closed, may be opened provided valve 1CS-510 is closed. |
| 1CS-93 (CS-D51SN) | Resin Sluice to CVCS Demineralizers | Lock closed. |
| 1CS-320 (CS-D641SN) | Boron Recycle Evaporator Feed Pump to Charging/SI Pumps | Lock closed. |

TABLE 3.9-1 (Continued)

ADMINISTRATIVE CONTROLS
TO PREVENT DILUTION DURING REFUELING

| CP&L VALVE NO. (Ebasco Valve No.) | DESCRIPTION | REQUIRED POSITION |
|---|---|---|
| 1CS-570 (CS-D575SN) | CVCS Letdown to Boron Thermal Regeneration System | Closed with main control board control switch in "shut" position, and BTRS function selector switch maintained in "off" position; no lock required. |
| 1CS-670 (CS-D599SN) | Reactor Makeup Water to Boron Thermal Regeneration System | Lock closed. |
| 1CS-649 (CS-D198SN) | Resin Sluice to BTRS Demineralizers | Lock closed. |
| 1CS-98 (CS-D740SN) | Boron Thermal Regeneration System Bypass | Opened with main control board control switch maintained in "open" position; no lock required. |

## 3/4.9.2 INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors* shall be OPERABLE, each with continuous visual indication in the control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, in addition to Action a. above, immediately initiate actions to restore one source range neutron flux monitor to OPERABLE status and determine the boron concentration of the Reactor Coolant System within 4 hours and once per 12 hours thereafter.

### SURVEILLANCE REQUIREMENTS

4.9.2 Each neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at the frequency specified in the Surveillance Frequency Control Program,

b. A CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program.

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* Wide Range Neutron Flux Monitors may be substituted for Source Range Neutron Flux Monitors provided the two required OPERABLE monitors (Source Range Neutron Flux Monitors and/or Wide Range Neutron Flux Monitors) are located on opposite sides of the core.

3/4.9.3 DECAY TIME - DELETED

## 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

### LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door is capable of being closed and held in place by a minimum of four bolts*,

b. A minimum of one door in each airlock is capable of being closed*, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Be capable of being* closed by a manual or automatic isolation valve, blind flange or equivalent, or

2. Be capable of being closed by OPERABLE automatic normal containment purge and containment pre-entry purge makeup and exhaust isolation valves*.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

### SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition, capable of being closed/isolated*, or capable of being closed by OPERABLE automatic normal containment purge and containment pre-entry purge makeup and exhaust isolation valves at the frequency specified in the Surveillance Frequency Control Program during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are either closed/isolated or capable of being closed/isolated*, or

b. Testing the normal containment purge and containment pre-entry purge makeup and exhaust isolation valves per the applicable portions of Specification 4.6.3.2.

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*Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be opened under administrative controls.

3/4.9.5 COMMUNICATIONS - DELETED

3/4.9.6 REFUELING MACHINE - DELETED

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING - DELETED

REFUELING OPERATIONS

## 3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

## HIGH WATER LEVEL

### LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, with irradiated fuel in the vessel when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

### SURVEILLANCE REQUIREMENTS

4.9.8.1.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2500 gpm at the frequency specified in the Surveillance Frequency Control Program.

4.9.8.1.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

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*The RHR loop may be removed from operation for up to 1 hour per 2-hour period during the performance of CORE ALTERATIONS and core loading verification in the vicinity of the reactor vessel hot legs.

## LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6, with irradiated fuel in the vessel when the water level above the top of the reactor 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 or to establish greater than or equal to 23 feet of water above the reactor vessel flange 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. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

## SURVEILLANCE REQUIREMENTS

4.9.8.2.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2500 gpm at the frequency specified in the Surveillance Frequency Control Program whenever the water level is at or above the reactor vessel flange.

4.9.8.2.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 900 gpm at the frequency specified in the Surveillance Frequency Control Program whenever the water level is below the reactor vessel flange.

4.9.8.2.3 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

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*The operating RHR loop may be removed from operation for up to 1 hour per 2-hour period during the performance of CORE ALTERATIONS and core loading verification in the vicinity of the reactor vessel hot legs.

## 3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Ventilation Isolation System inoperable, close each of the containment purge makeup and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere.*

b. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at the frequency specified in the Surveillance Frequency Control Program during CORE ALTERATIONS by verifying that containment ventilation isolation occurs on a two-out-of-four High Radiation test signal from the containment area radiation monitors (Table 3.3-6, item 1.a) and by verifying that isolation occurs for each valve using its control switch in the main control room.

*Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be opened under administrative controls.

## 3/4.9.10 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 vessel flange.

APPLICABILITY: MODE 6, during movement of irradiated fuel assemblies within containment, or during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts.

ACTION:

With the requirements of the above specification not satisfied, suspend CORE ALTERATIONS, including operations involving movement of fuel assemblies within containment, and initiate actions to restore refueling cavity water level to within limits.

### SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth at the frequency specified in the Surveillance Frequency Control Program.

## 3/4.9.11 WATER LEVEL – NEW AND SPENT FUEL POOLS

### LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of fuel rods within irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in a pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the affected pool area and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.11 At the frequency specified in the Surveillance Frequency Control Program, when irradiated fuel assemblies are in a pool, the water level in that pool shall be determined to be at least its minimum required depth.

## 3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

### LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Handling Building Emergency Exhaust System Trains shall be OPERABLE.*

APPLICABILITY: Whenever irradiated fuel is in a storage pool.

ACTION:

a. With one Fuel Handling Building Emergency Exhaust System Train inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Handling Building Emergency Exhaust System Train is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorber.

b. With no Fuel Handling Building Emergency Exhaust System Trains OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Handling Building Emergency Exhaust System Train is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Emergency Exhaust System trains shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 continuous minutes;

b. At the frequency specified in the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

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* The Fuel Handling Building Emergency Exhaust System boundary may be opened intermittently under administrative controls.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.9.12 (Continued)

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 95% in accordance with ASTM D3803-1989.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodide penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 95% in accordance with ASTM D3803-1989.

d. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber bank is not greater than 4.1 inches water gauge while operating the unit at a flow rate of 6600 cfm ± 10%,

2. Verifying that, on a High Radiation test signal, the system automatically starts and directs its exhaust flow through the HEPA filters and charcoal adsorber banks,

3. Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inch water gauge, relative to the outside atmosphere, during system operation at a flow rate of 6600 cfm ± 10%,and

4. Deleted

5. Deleted.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the unit at a flow rate of 6600 cfm ± 10%.

FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.9.12 (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the unit satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 6600 cfm ± 10%.

## 3/4.10 SPECIAL TEST EXCEPTIONS

## 3/4.10.1 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 shutdown and control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated single rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any shutdown and control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all shutdown and control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 shutdown and control rod either partially or fully withdrawn shall be determined at the frequency specified in the Surveillance Frequency Control Program.

4.10.1.2 Each shutdown and control 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

3/4.10.2 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, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 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 Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 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 less than or equal to 85% of RATED THERMAL POWER at the frequency specified in the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at the frequency specified in the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Specification 4.2.2.2 and

b. Specification 4.2.3.2.

## 3/4.10.3 PHYSICS TESTS

### LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature ($T_{avg}$) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

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

### SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at the frequency specified in the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature ($T_{avg}$) shall be determined to be greater than or equal to 541°F at the frequency specified in the Surveillance Frequency Control Program during PHYSICS TESTS.

## 3/4.10.4 REACTOR COOLANT LOOPS

### 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 in MODE 1 or 2 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 less than or equal to 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 at the frequency specified in the Surveillance Frequency Control Program during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST prior to initiating startup and PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

## 3/4.10.5 POSITION INDICATION SYSTEM – SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

### SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at the frequency specified in the Surveillance Frequency Control Program thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) $k_{eff}$ is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

## 3/4.11 RADIOACTIVE EFFLUENTS

### 3/4.11.1 LIQUID EFFLUENTS

Specifications 3/4.11.1.1, 3/4.11.1.2 and 3/4.11.1.3 have been deleted from Technical Specifications and relocated to the ODCM.

Pages 3/4 11-2 through 3/4 11-6 have been deleted.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS - DELETED

## 3/4.11.2 GASEOUS EFFLUENTS

Specifications 3/4.11.2.1, 3/4.11.2.2, 3/4.11.2.3 and 3/4.11.2.4 have been deleted from Technical Specifications and relocated to the ODCM.

Pages 3/4 11-9 through 3/4 11-14 have been deleted.

EXPLOSIVE GAS MIXTURE - DELETED

GAS STORAGE TANKS - DELETED

## 3/4.11.3 SOLID RADIOACTIVE WASTES

Specification 3/4.11.3 has been deleted from Technical Specifications and relocated to the PCP.

Page 3/4 11-18 has been deleted.

## 3/4.11.4 TOTAL DOSE

Specification 3/4.11.4 has been deleted from Technical Specifications and relocated to the ODCM.

## 3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

Specifications 3/4.12.1, 3/4.12.2 and 3/4.12.3 have been deleted from Technical Specifications and relocated to the ODCM.

Pages 3/4 12-2 through 3/4 12-14 have been deleted.

# SECTION 5.0

# DESIGN FEATURES

## 5.0 DESIGN FEATURES

### 5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area Boundary shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4.

### 5.2 CONTAINMENT

CONFIGURATION

5.2.1 The 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 = 130 feet.

b. Nominal inside height = 160 feet from the liner on the foundation mat to the spring line, 225 feet from the liner on the foundation mat to the dome peak.

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete dome = 2.5 feet.

e. Minimum thickness of concrete floor pad over the containment liner = 5.0 feet.

f. Nominal thickness of steel liner = 0.375 inches in the cylindrical portion, 0.25 inches on the bottom, and 0.5 inches in the dome.

g. Net free volume = $2.266 \times 10^6$ cubic feet.

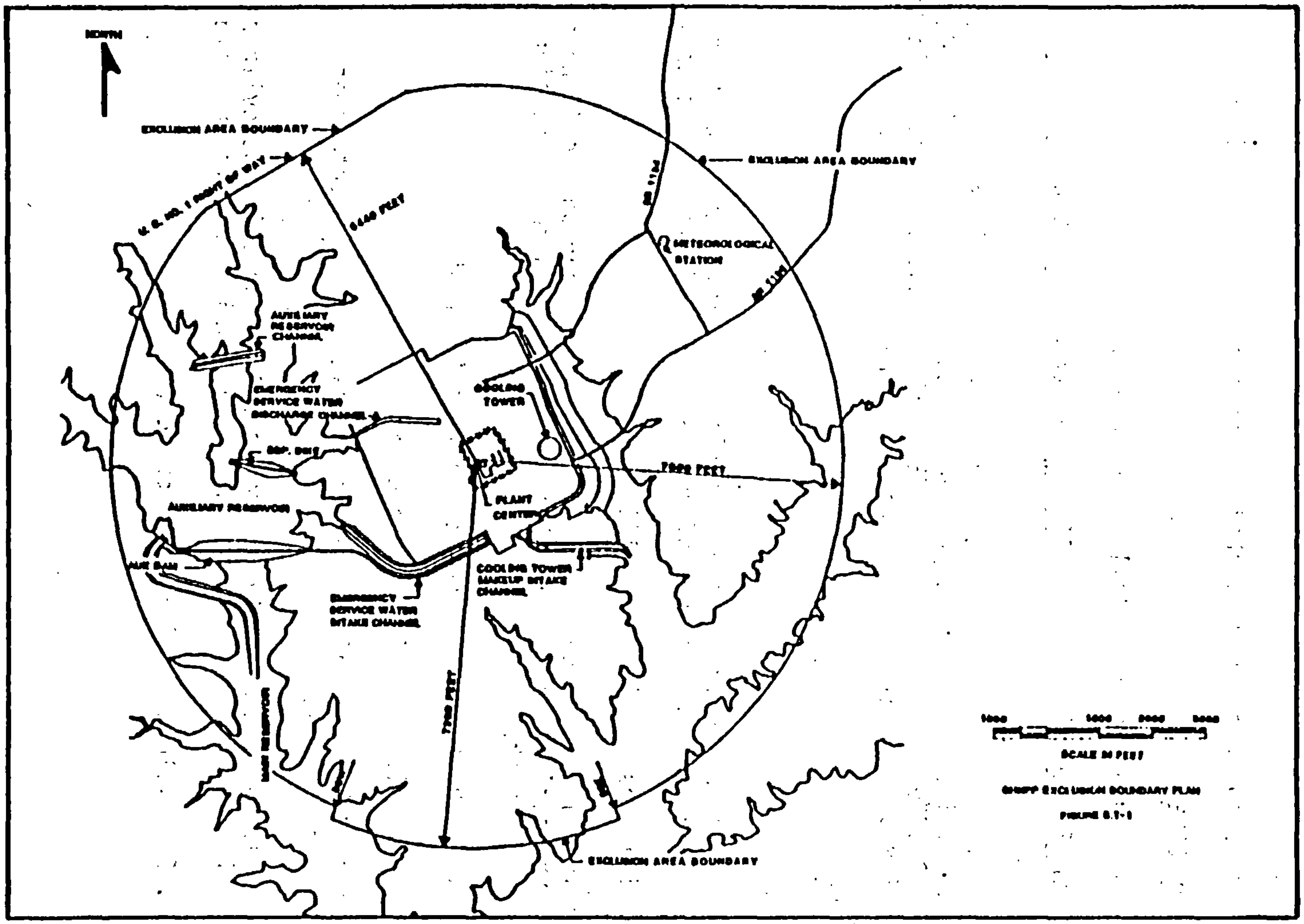



FIGURE 5.1-1

EXCLUSION AREA

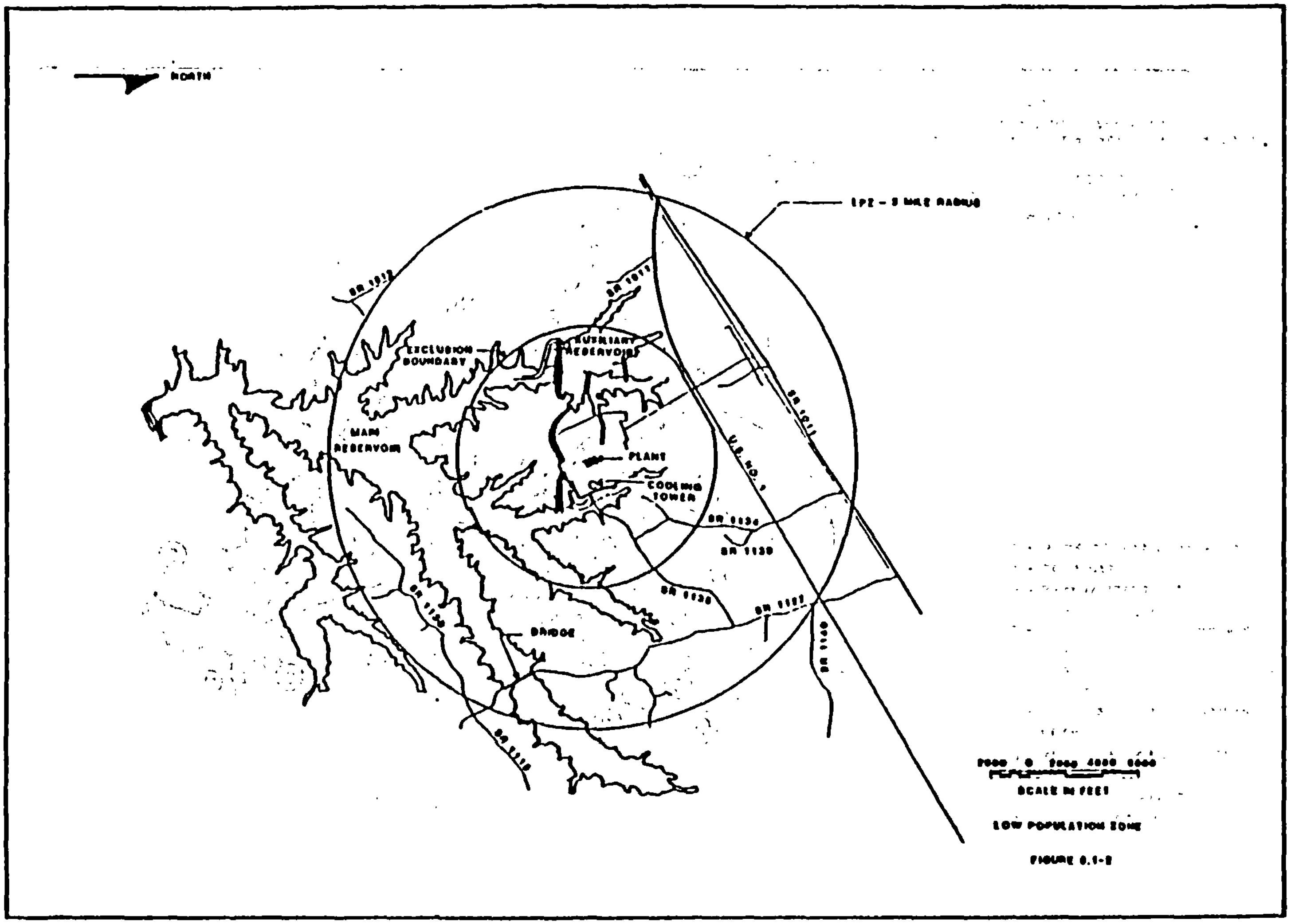



FIGURE 5.1-2

LOW POPULATION ZONE

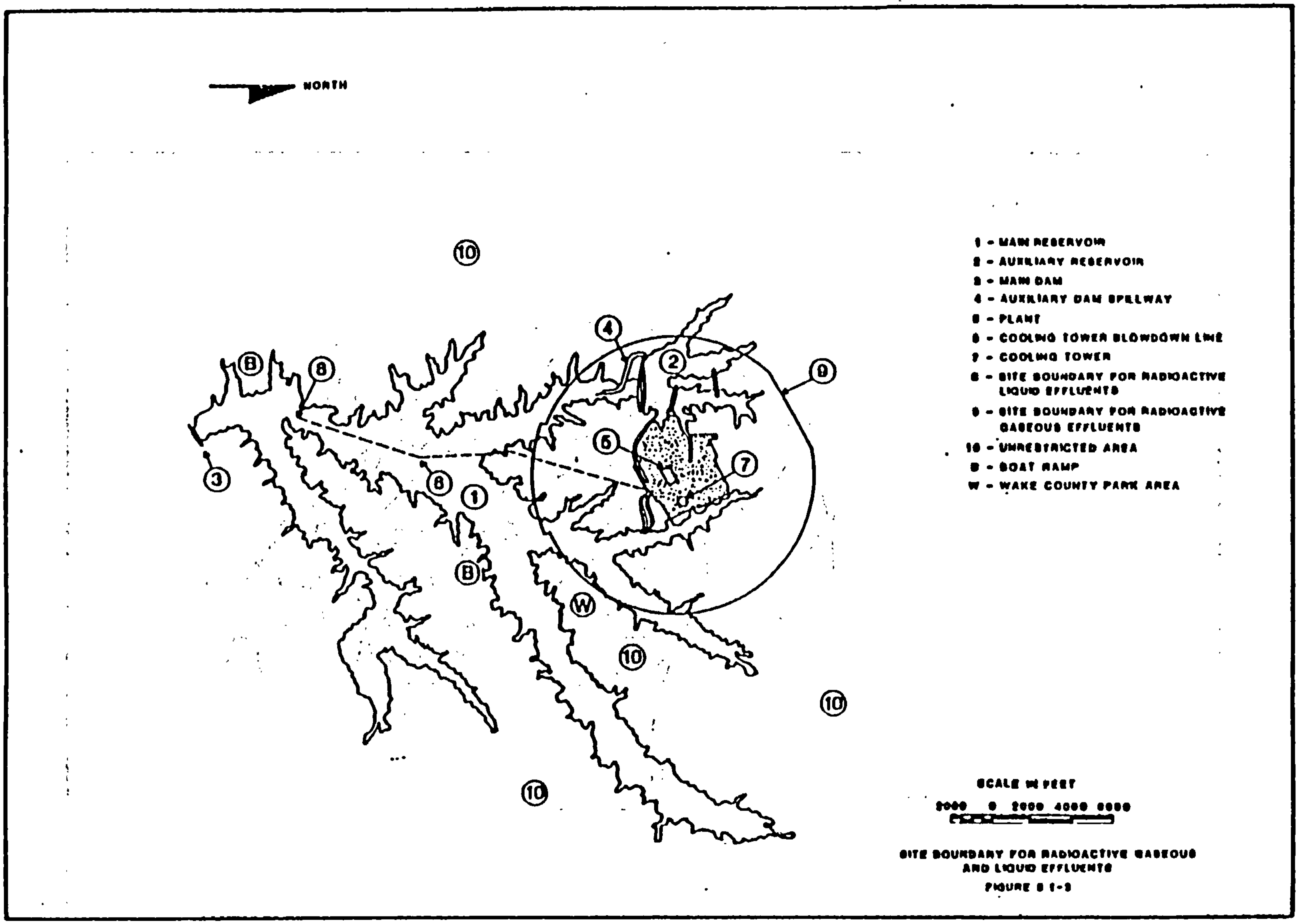



FIGURE 5.1-3

SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

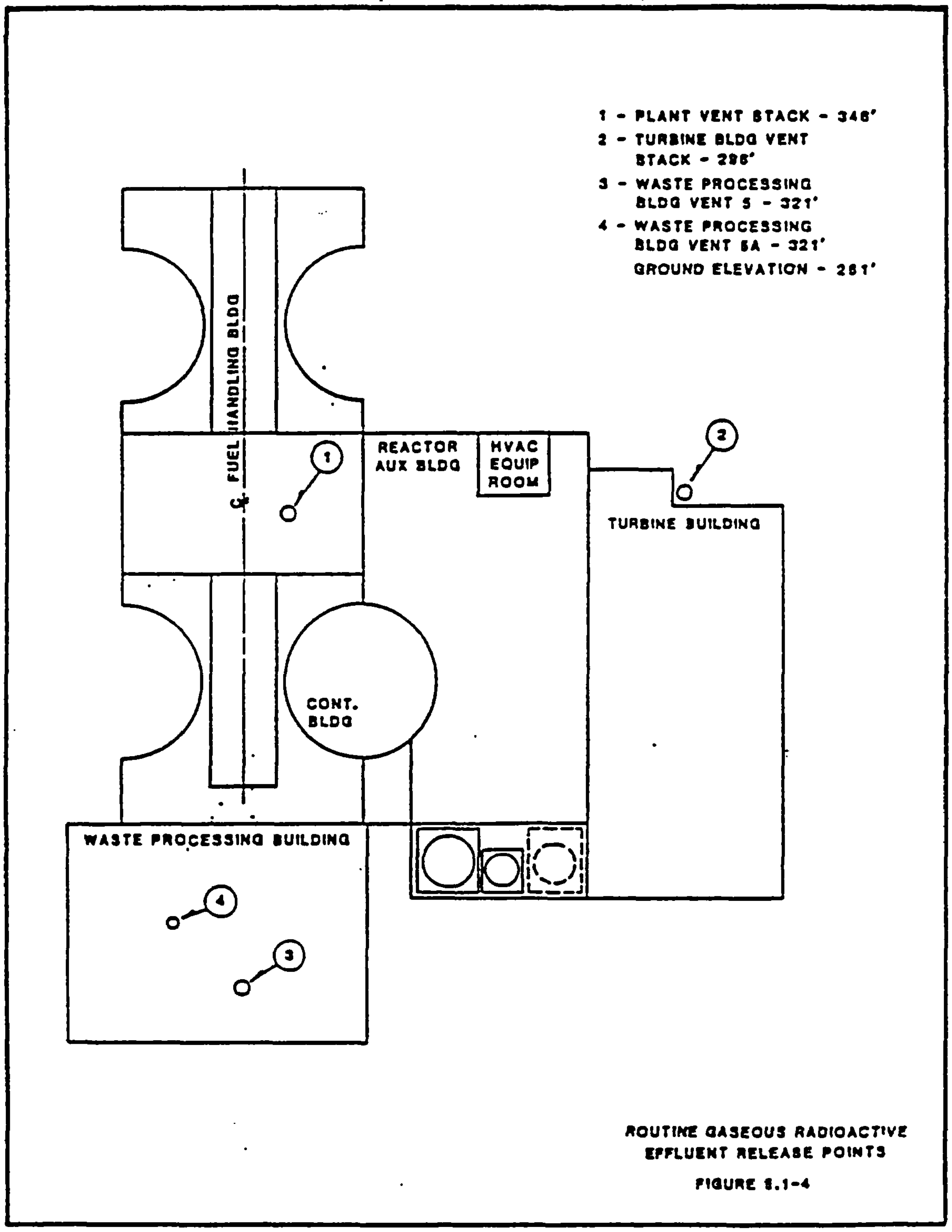



FIGURE 5.1-4

ROUTINE GASEOUS RADIOACTIVE EFFLUENT RELEASE POINTS

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

## 5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 or M5. Limited substitution of fuel rods by filler rods (consisting of Zircaloy-4 or M5 clad stainless steel or zirconium), or vacancies may be made in fuel assemblies if justified by a cycle specific evaluation. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U 235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. 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 5.2 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.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is approximately 10,300 cubic feet at a nominal $T_{avg}$ of 588.8°F.

## 5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological station shall be located as shown on Figure 5.1-1.

## 5.6 FUEL STORAGE

### CRITICALITY

5.6.1 The fuel storage racks are designed and shall be maintained with:

1. PWR storage racks in pools "A" and "B"

   a. $k_{eff}$ less than or equal to 0.95 if fully flooded with water borated to 2000ppm.

   b. $k_{eff}$ less than 1.0 if flooded with unborated water.

   c. A nominal 10.5 inch center-to-center distance between fuel assemblies.

   d. Assemblies must be within the "acceptable range" of the burnup restrictions shown in Figure 5.6-2 prior to storage in unrestricted storage.

   e. Assemblies that do not meet the requirements of 5.6.1.1.d shall be stored in a 2-of-4 checkerboard within and across rack module boundaries. Less dense storage patterns (e.g. 1-of-4 or 1-of-5) are acceptable in place of 2-of-4.

   f. The empty spaces (water holes) in the 5.6.1.1.e checkerboard may be occupied by non-fuel items (e.g., containment specimen and trash baskets, mock fuel assemblies etc.) up to a limit of one per every 6 storage spaces.

   g. If fuel that meets the requirement of 5.6.1.1.d and fuel that does not meet 5.6.1.1.d are stored in the same rack module, an interface region must exist between the two regions. The interface region shall either be an empty row/column or a row/column of fuel that meets the requirements of 5.6.1.1.d in a checkerboard pattern with the restricted (5.6.1.1.e) region.

2. Dry New Fuel PWR Storage Racks

   a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.

   b. $k_{eff} \leq 0.95$ if fully flooded with unborated water without credit for Boraflex in the rack module.

   c. $k_{eff} \leq 0.98$ in an optimum moderation event.

   d. A nominal 10.5 inch center to center distance between storage cells with alternating rows and columns blocked such that fuel is stored in a 1-of-4 pattern.

3. BWR Storage Racks in Pools "A" and "B"

   a. Racks with Metamic neutron absorber inserts

      1. $k_{eff}$ less than or equal to 0.95 when flooded with water borated to 1000 ppm.
      2. $k_{eff}$ less than 1.0 when flooded with unborated water.
      3. The reactivity margin is assured for BWR racks in pools “A” and “B” by maintaining a nominal 6.25 inch center-to-center distance in the BWR storage racks.
      4. The following restrictions are also imposed through administrative controls:

         a. Storage of BWR fuel designs limited to GE3, GE4, GE5, GE6, and GE7 fuel designs.
         b. Rack insert orientation is limited to that shown in Figure 5.6-3 and Figure 5.6-4.
         c. No fuel shall be stored in Storage Location A11 of Rack C1 in Spent Fuel Pool A.

   b. Racks with Boral neutron absorber

      1. $k_{eff}$ less than or equal to 0.95 when flooded with unborated water.
      2. The reactivity margin is assured for BWR racks in pools “A” and “B” by maintaining a nominal 6.25 inch center-to-center distance in the BWR storage racks.

4. PWR and BWR racks in pools "C" and "D"

   a. $k_{eff}$ less than or equal to 0.95 when flooded with unborated water.
   b. The reactivity margin is assured for pools "C" and "D" by maintaining a nominal 9.017 inch center-to-center distance between fuel assemblies placed in the non-flux trap style PWR storage racks and 6.25 inch center-to-center distance in the BWR storage racks.
   c. The following restrictions are also imposed through administrative controls:

      1. PWR assemblies must be within the "acceptable range" of the burnup restrictions shown in Figure 5.6-1 prior to storage in pools "C" and "D".
      2. BWR assemblies are acceptable for storage in pool "C" provided the maximum planar average enrichments are less than 4.6 wt.% U235 and $K_{inf}$ is less than or equal to 1.32 for the standard cold core geometry (SCCG).

5. In each case, $k_{eff}$ includes allowances for uncertainties as described in Section 4.3.2.6 of the FSAR.

DRAINAGE

5.6.2 The pools "A", "B", "C" and "D" are designed and shall be maintained to prevent inadvertent draining of the pools below elevation 277.

CAPACITY

5.6.3.a Pool "A" contains six (6 x 10 cell) flux trap type PWR racks and three (11 x 11 cell) BWR racks for a total storage capacity of 723 assemblies. Pool "B" contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) flux trap style PWR racks and seventeen (11 x 11 cell) BWR racks and is licensed for one additional (11 x 11 cell) BWR rack that will be installed as needed. The combined pool "A" and "B" licensed storage capacity is 3669 assemblies.

5.6.3.b Pool "C" is designed to contain a combination of PWR and BWR assemblies. Pool "C" can contain two (11 x 9 cell) and thirteen (9 X 9 cell) PWR racks for storage of 1251 PWR assemblies. Pool "C" can contain two (8 x 13 cell), two (8 x 11 cell), six (13 x 11 cell), and five (13 x 13 cell) BWR racks for storage of 2087 BWR assemblies. The (9 x 9 cell) PWR racks and the (13 x 13 cell) BWR racks are dimensioned to allow interchangeability between PWR or BWR storage rack styles as required. The racks in pool "C" will be installed as needed.

5.6.3.c Pool "D" contains a variable number of PWR storage spaces. These racks will be installed as needed. Pool "D" is designed for a maximum storage capacity of 1025 PWR assemblies.

5.6.3.d The heat load from fuel stored in Pools "C" and "D" shall not exceed 7.0 MBtu/hr.

## 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.

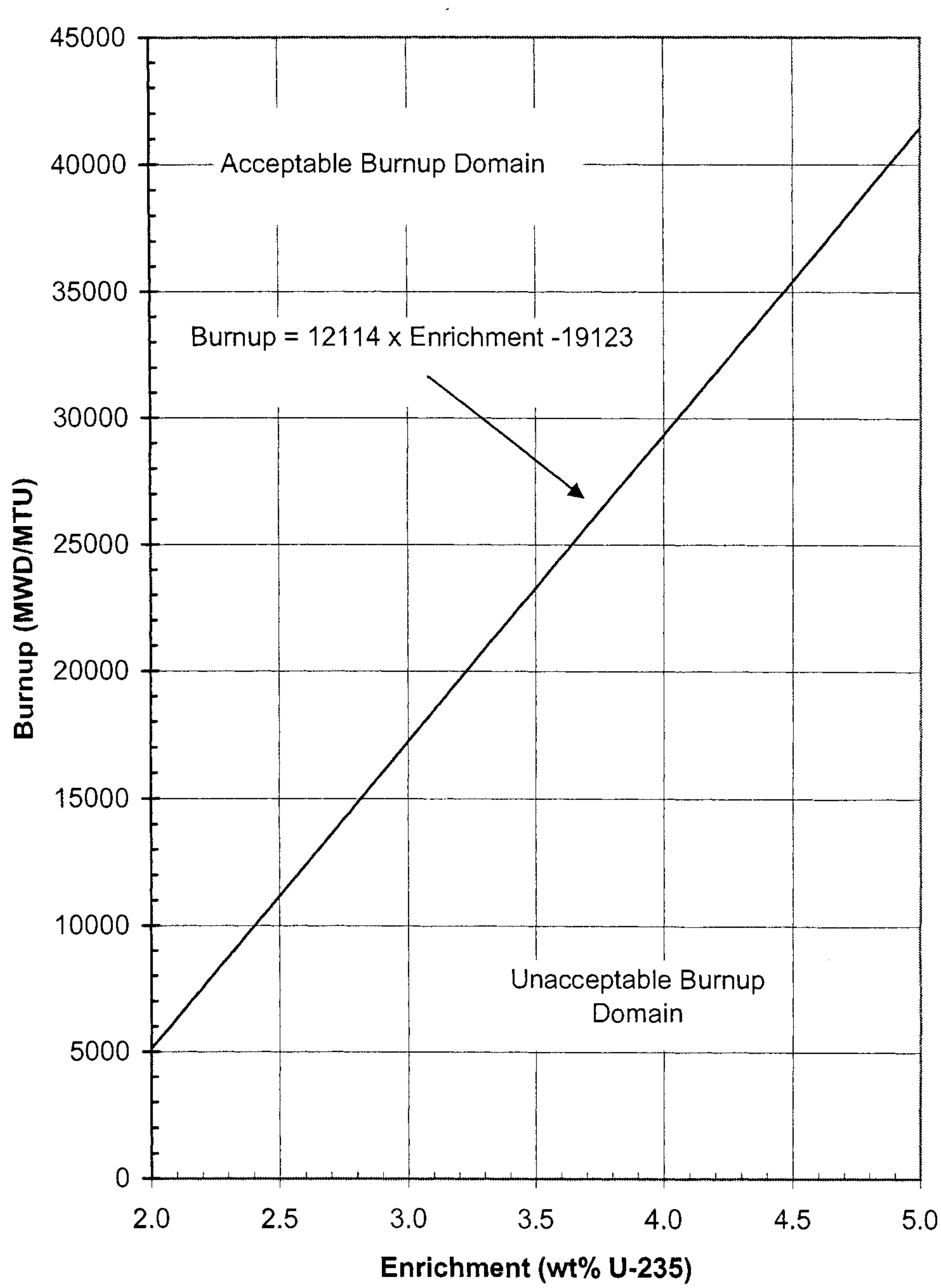



FIGURE 5.6-1
POOLS "C" and "D" BURNUP VERSUS ENRICHMENT FOR PWR FUEL

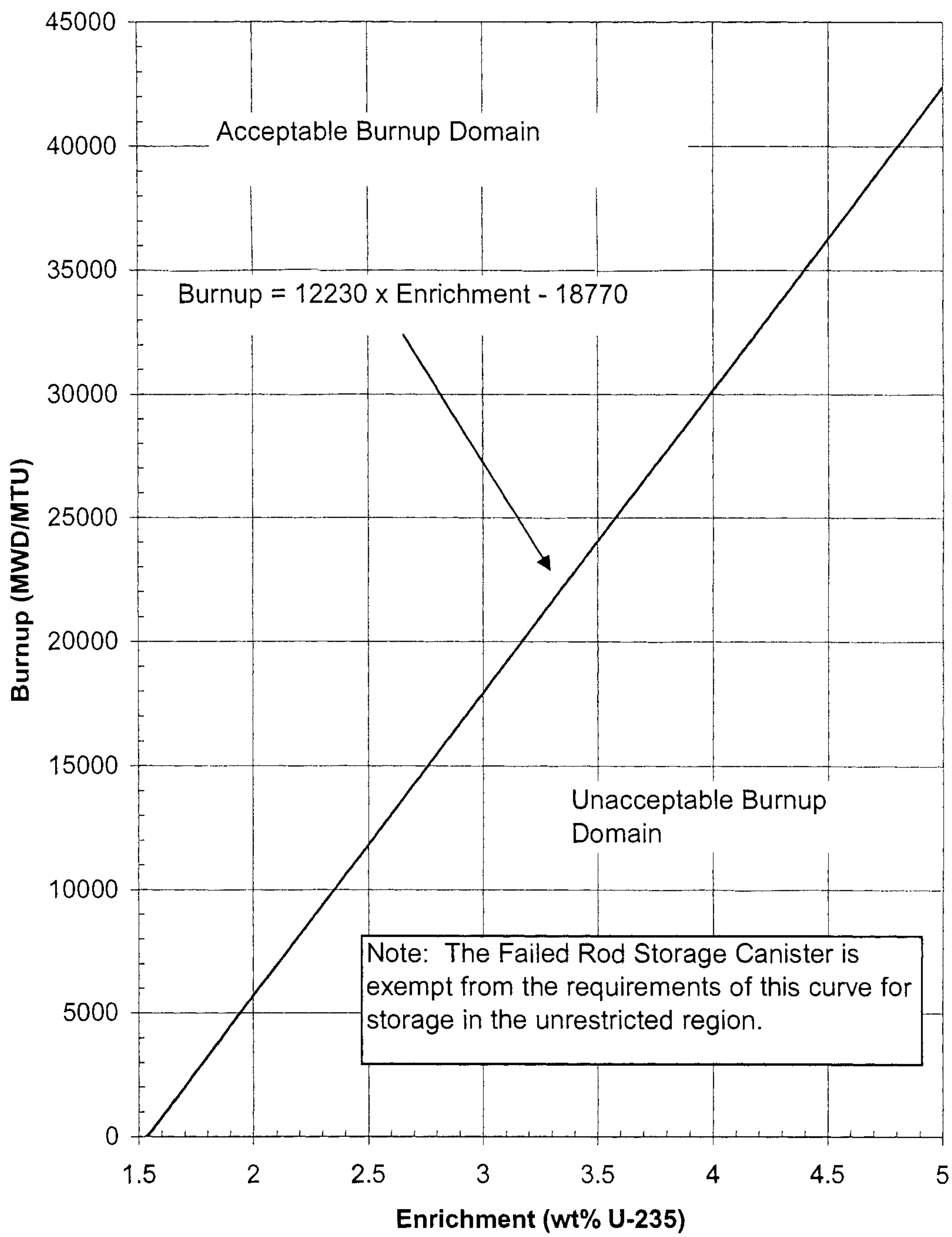



FIGURE 5.6-2
POOLS "A" and "B" BURNUP VERSUS ENRICHMENT FOR PWR FUEL

<table>
<tr><td colspan="4">PWR Racks</td></tr>
<tr><td>BWR Rack With Inserts</td><td>BWR Rack With Inserts</td><td>BWR Rack With Inserts</td><td>PWR Racks</td></tr>
</table>

Plant North 

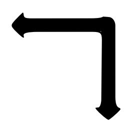
Insert Orientation 

FIGURE 5.6-3

POOL A METAMIC RACK INSERT ORIENTATION

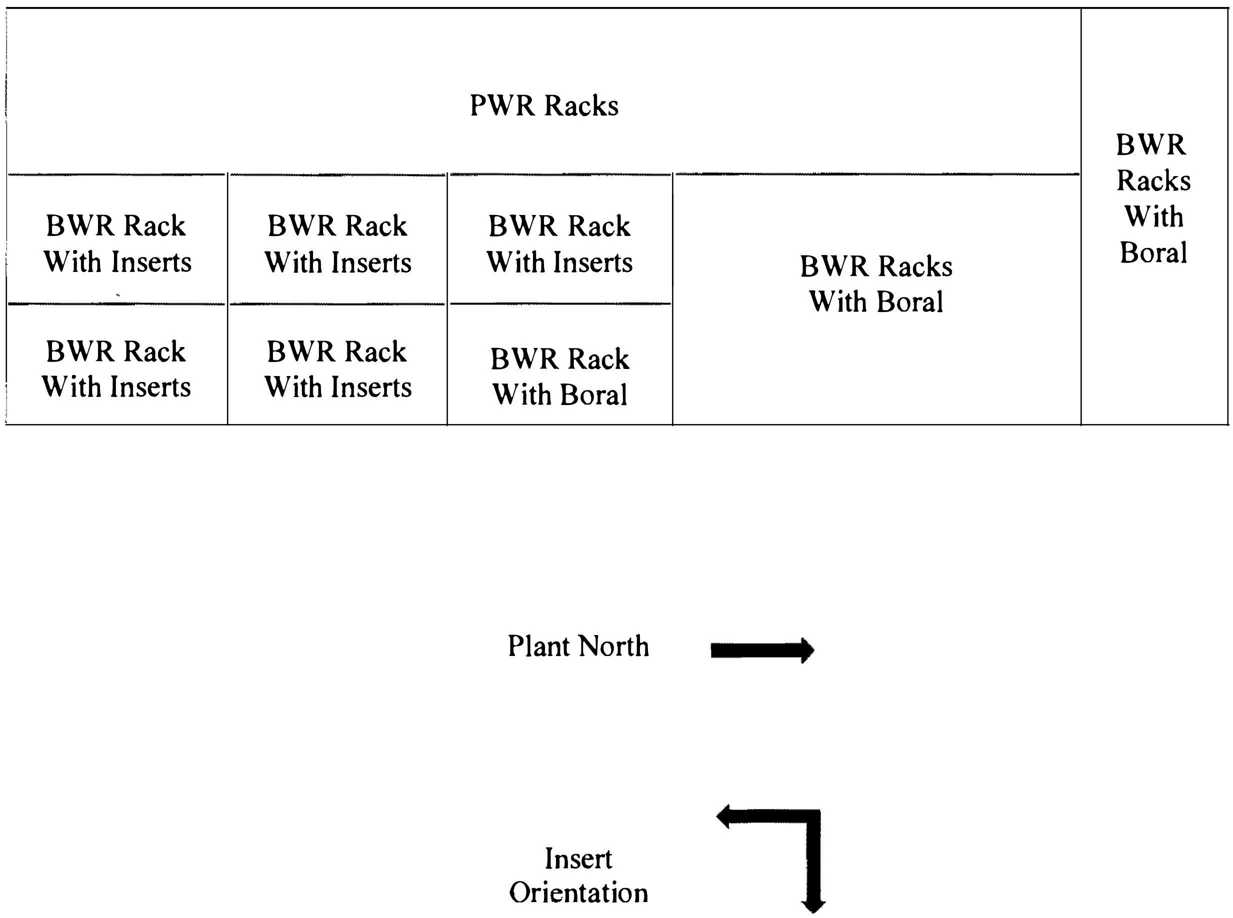



FIGURE 5.6-4

POOL B METAMIC RACK INSERT ORIENTATION

## 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/h and 200 cooldown cycles at $\leq$ 100°F/h. | Heatup cycle - $T_{avg}$ from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - $T_{avg}$ from $\geq$ 550°F to $\leq$ 200°F. |
| | 200 pressurizer cooldown cycles at $\leq$ 200°F/h. | Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F. |
| | 200 loss of load cycles, without immediate Turbine or Reactor trip. | $\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER. |
| | 40 cycles of loss-of-offsite A.C. electrical power. | Loss-of-offsite A.C. electrical ESF Electrical 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. |
| | 10 auxiliary spray actuation cycles. | Spray water temperature differential greater than 320°F but less than 625°F. |
| | 200 leak tests. | Pressurized to $\geq$ 2485 psig. |
| | 10 hydrostatic pressure tests. | Pressurized to $\geq$ 3107 psig. |
| Secondary Coolant System | 1 steam line break. | Break in a > 6-inch steam line. |
| | 10 hydrostatic pressure tests. | Pressurized to $\geq$ 1481 psig. |

# SECTION 6.0

# ADMINISTRATIVE CONTROLS

## 6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Superintendent-Shift Operations (or, during his absence from the control room, a designated individual) shall be responsible for the control room command function.

## 6.2 ORGANIZATION

### 6.2.1 Onsite And Offsite Organization

An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the 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. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the FSAR and updated in accordance with 10 CFR 50.71(e).

b. There shall be an individual executive position (corporate officer) in the offsite organization having corporate responsibility for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. There shall be an individual management position in the onsite organization having responsibility for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the health physics manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

UNIT STAFF

6.2.2 The unit organization shall be subject to the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. An individual qualified as a Radiation Control Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. The Manager-Operations shall meet one of the following:

(1) Hold a Senior Operator License or,

(2) Have held a Senior Operator License for a similar unit, or

(3) Have been certified for equivalent senior operator knowledge for a similar unit.

If the Manager-Operations does not hold a Senior Reactor Operator License, an off-shift Operations superintendent who reports directly to the Manager-Operations and holds a Senior Reactor Operator License shall be designated to supervise shift work and licensed activities.

* The Radiation Control Technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

f. Deleted by Amendment No. 130

FIGURE 6.2-1

DELETED

FIGURE 6.2-2

DELETED

## TABLE 6.2-1

## MINIMUM SHIFT CREW COMPOSITION

| POSITION | NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION | |
|---|---|---|
| | MODE 1, 2, 3, or 4 | MODE 5 or 6 |
| S-SO | 1 | 1 |
| SRO | 1 | None |
| RO | 2 | 1 |
| AO | 2 | 1 |
| STA | 1* | None |

S-SO - Superintendent-Shift Operations with a Senior Operator license on Unit 1
SRO - Individual with a Senior Operator license on Unit 1
RO - Individual with an Operator license on Unit 1
AO - Auxiliary Operator - license not required
STA - Shift Technical Advisor

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 unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Superintendent-Shift Operations from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Superintendent-Shift Operations from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

* The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Superintendent-Shift Operations or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.



Correction letter of 10-29-98

6.2.3 DELETED

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Superintendent-Shift Operations in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a baccalaureate degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 Deleted

## 6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained and shall meet or exceed the requirements referenced for comparable positions, as specified in the Duke Energy Corporation Quality Assurance Program Description (DUKE-QAPD-001-A).

## 6.5 DELETED

(PAGES 6-8 THROUGH 6-15 DELETED)

(NEXT PAGE IS 6-16)

## 6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the On-Site Review Committee (ORC), and the results of this review shall be submitted to the Manager - Nuclear Assessment Section and the Vice President - Harris Nuclear Plant.

## 6.7 SAFETY LIMIT VIOLATION

6.7.1 Deleted.

## 6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

PROCEDURES AND PROGRAMS (Continued)

g. Quality Assurance Program for effluent and environmental monitoring; and

h. Deleted.

i. Technical Specification Equipment List Program.

6.8.2 DELETED

6.8.3 DELETED

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage, to as low as practical levels, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident The systems include:

1. Residual Heat Removal System and Containment Spray System, except spray additive subsystem and RWST,

2. Safety Injection System, except boron injection recirculation subsystem and accumulator,

3. Portions of the Chemical and Volume Control System:

   a. Letdown subsystem, including demineralizers,
   b. Boron re-cycle holdup tanks, and
   c. Charging/safety injection pumps,

4. Post-Accident Sample System (until such time as a modification eliminates the Post-Accident Sample System as a potential leakage path).

PROCEDURES AND PROGRAMS (Continued)

a. Primary Coolant Sources Outside Containment (Continued)

5. Post-Accident Reactor Auxiliary Building Ventilation System,

6. Valve Leakoff Equipment Drain System,

7. Gaseous Waste Processing System, and

8. Seal Water Return System.

9. The portion of the Filter Backwash System that services the 'A' and 'B' reactor coolant pump seal injection filters.

The program shall include (1) preventive maintenance and periodic visual inspection requirements and (2) integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program that will ensure the capability to determine accurately the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical variables and the control points for these variables,

2. Identification of the procedures used to measure the values of the critical variables,

3. Identification of process sampling points, which shall include monitoring for evidence of condenser in-leakage,

4. Procedures for the recording and management of data,

5. Procedures defining corrective actions for all off-control point chemistry conditions, and

PROCEDURES AND PROGRAMS (Continued)

c. Secondary Water Chemistry (Continued)

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.

d. Backup Method for Determining Subcooling Margin

A program that will ensure the capability to monitor accurately the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and
2. Procedures for monitoring.

e. DELETED

f. Inspections of Water Control Structures

A program to implement an ongoing inspection program in accordance with Regulatory Guide 1.127 (Revision 1, March 1978) for the main and auxiliary dams, the auxiliary separating dike, the emergency service water intake and discharge channels, and the auxiliary reservoir channel. The program shall include the following:

1. The provisions of Regulatory Guide 1.127, Revision 1, to be implemented as a part of plant startup operations.

2. Subsequent inspections at yearly intervals for at least the next 3 years. If adverse conditions are not revealed by these inspections, inspection at 5-year intervals will be performed.

g. Turbine Rotor Inspection

A program to implement an ongoing inspection of the low pressure turbine rotor. The program shall be based upon:

1. Vendor recommendations for low pressure turbine rotor inspection intervals and procedural guidelines, and

2. Using vendor methodology to recalculate the inspection interval if cracking in the rotor is ever found.

PROCEDURES AND PROGRAMS (Continued)

h. 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 MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

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 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC 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 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

   a. For noble gases: less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

PROCEDURES AND PROGRAMS (Continued)

h. Radioactive Effluent Controls Program (Cont.)

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents 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 to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

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.

i. 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 this 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.

j. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System (GWPS), the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

The program shall include:

1) The limits for concentrations of hydrogen and oxygen in the GWPS and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion).

PROCEDURES AND PROGRAMS (Continued)

j. Explosive Gas and Storage Tank Radioactivity Monitoring Program (Cont.)

2) A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

3) A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Processing System is less than the amount that would result in concentrations that exceed the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

PROCEDURES AND PROGRAMS (Continued)

k. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,” Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, with the following exception noted:

> Visual examination of the containment system shall be in accordance with Specification 4.6.1.6.1.

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident is 41.8 psig. The calculated peak containment internal pressure related to the design basis main steam line break is 41.3 psig. $P_a$ will be assumed to be 41.8 psig for the purpose of containment testing in accordance with this Technical Specification.

The maximum allowable containment leakage rate, $L_a$ at $P_a$, shall be 0.1 % of containment air weight per day.

Leakage rate acceptance criteria:

1) The containment leakage rate acceptance criterion is ≤ 1.0 $L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 $L_a$ for the combined Type B and Type C tests, and ≤ 0.75 $L_a$ for Type A tests.

2) Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is ≤ 0.05 $L_a$ when tested at ≥ $P_a$.

   b) For each door, leakage rate is ≤ 0.01 $L_a$ when pressurized to ≥ $P_a$.

The provisions of Surveillance Requirement 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing frequencies required by 10 CFR 50, Appendix J.

PROCEDURES AND PROGRAMS (Continued)

l. 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:

1. 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 a 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.

2. Performance criteria for SG tube integrity. Steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   a) Structural integrity performance criterion. All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, HOT STANDBY, and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 (3deltaP) 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.

   b) 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. Accident induced leakage is not to exceed 1 gpm total for all three SGs.

   c) The operation leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

PROCEDURES AND PROGRAMS (Continued)

3. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

4. 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 tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of 4a, 4b, and 4c 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.

   a) Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

   b) After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

   c) If crack indications are found in any SG tube, 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. 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.

5. Provisions for monitoring operational primary-to-secondary leakage.

(PAGE 6-19f DELETED By Amendment No. 191)

PROCEDURES AND PROGRAMS (Continued)

m. Inservice Testing Program (Deleted)

Note: See Section 1.17a for the definition of INSERVICE TESTING PROGRAM.

**PROCEDURES AND PROGRAMS (Continued)**

n. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a. A change in the TS incorporated in the license, or

   b. A change to the FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.n.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

o. CONTROL ROOM ENVELOPE HABITABILITY PROGRAM

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System (CREFS), 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 exposure in excess of 5 rem TEDE, or its equivalent, for the duration of the accident. The program shall include the following elements:

1. The definition of the CRE and the CRE boundary.

2. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

3. 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.

4. 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 subsystem of the CREFS, operating at the flow rate required by SR 4.7.6.d.1, at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the assessment of the CRE boundary required by paragraph 3, requirement (ii).

5. The quantitative limits on unfiltered air leakage 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 3. 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.

6. The provisions of Surveillance Requirement 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 3 and 4, respectively.

PROCEDURES AND PROGRAMS (Continued)

p. 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.

q. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

1. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   a. An API gravity or an absolute specific gravity within limits,
   b. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
   c. A clear and bright appearance with proper color or a water and sediment content within limits.

2. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in 1., above, are within limits for ASTM 2D fuel oil, and

3. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of Surveillance Requirement 4.0.2 and Surveillance Requirement 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

PROCEDURES AND PROGRAMS (Continued)

r. Risk-Informed Completion Time Program

This program provides controls to calculate a Risk-Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1 and 2;

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

   1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

   2. For emergent conditions, the revised RICT must be determined within the time limits of the Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

   3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

   1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

   2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

## 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 NRC in accordance with 10CFR50.4.

6.9.1.1 Deleted.

6.9.1.2 Deleted.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.4 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to 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 objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

6.9.1.5 Deleted

## 6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.1 and 3/4.1.1.2.

b. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.

c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.

d. Control Bank Insertion Limits for Specification 3/4.1.3.6.

e. Axial Flux Difference Limits for Specification 3/4.2.1.

f. Heat Flux Hot Channel Factor $F_{Q}(X,Y,Z)$ Limits for Specification 3/4.2.2.

g. Enthalpy Rise Hot Channel Factor $F_{\Delta H}(X,Y)$ Limits for Specification 3/4.2.3.

h. Boron Concentration for Specification 3/4.9.1.

i. Reactor Core Safety Limits Figure for Specification 2.1.1.

j. Overtemperature ΔT and Overpower ΔT setpoint parameters and time constant values for Specification 2.2.1.

k. Reactor Coolant System pressure, temperature, and flow Departure from Nucleate Boiling (DNB) limits for Specification 3/4.2.5.

l. Shutdown and Operating Boric Acid Tank and Refueling Water Storage Tank boron concentration limits for Specification 3/4.1.2.5 and 3/4.1.2.6.

m. ECCS Accumulators and Refueling Water Storage Tank boron concentration limits for Specification 3/4.5.1 and 3/4.5.4.

6.9.1.6.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and the approved revision number shall be identified in the COLR.

a. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

b. EMF-2103(P)(A), “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” approved version as specified in COLR.

c. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

d. BAW-10240(P)(A), “Incorporation of M5 Properties in Framatome ANP Approved Methods.”

e. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.

f. Mechanical Design Methodologies

BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” approved version as specified in the COLR.

g. DPC-NE-2005-P-A, “Thermal-Hydraulic Statistical Core Design Methodology,” approved version as specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

h. DPC-NE-1008-P-A, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,” as approved by NRC Safety Evaluation dated May 18, 2017.

i. DPC-NF-2010-A, “Nuclear Physics Methodology for Reload Design,” as approved by NRC Safety Evaluation dated May 18, 2017.

j. DPC-NE-2011-P-A, “Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors” as approved by NRC Safety Evaluation dated May 18, 2017.

k. DPC-NE-3008-P-A, “Thermal-Hydraulic Models for Transient Analysis,” as approved by NRC Safety Evaluation dated April 10, 2018.

l. DPC-NE-3009-P-A, “FSAR / UFSAR Chapter 15 Transient Analysis Methodology,” as approved by NRC Safety Evaluation dated April 10, 2018.

m. ANP-10341P-A, “The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations,” approved version as specified in the COLR.

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 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 Specification 6.8.4.l. The report shall include:

a. The scope of inspections performed on each SG;

b. The 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; and

   4. The number of tubes plugged during the inspection outage.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT (Continued)

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; and

f. The results of any SG secondary side inspections.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 DELETED

(PAGES 6-24c THROUGH 6-24d DELETED By Amendment No. 185)

(PAGE 6-25 DELETED By Amendment No.92)

## 6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

## 6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Sources or from any Surface Penetrated by the Radiation

a. Each accessible 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 a 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:

   1. Possess a radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"); or

   2. Possess 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 ("alarming dosimeter"), with an appropriate alarm setpoint; or

   3. Possess 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. Possess a self-reading dosimeter and 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; or

HIGH RADIATION AREA (Continued)

5. 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 and indicating device; who is responsible for controlling personnel radiation exposure within 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 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 accessible entryway to such an area shall be conspicuously posted as a locked 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 Superintendent - Shift Operations or the Radiation Control Supervisor or designated representative; and

2. Doors and gates shall remain locked or guarded except during periods of personnel or eqiupment 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:

1. Possess an alarming dosimeter with an appropriate alarm setpoint; or

2. Possess 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

HIGH RADIATION AREA (Continued)

3. Possess a direct-reading dosimeter and 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; or

4. 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 and indicating device; who is responsible for controlling personnel exposure within the area; or

5. In those cases where the options of Specifications 6.12.2.d.2, 6.12.2.d.3, and 6.12.2.d.4 above, are impractical or determined to be inconsistent with the "As Low As Reasonably Achievable" principle, possess a radiation monitoring and indicating device.

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 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 and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

HIGH RADIATION AREA (Continued)

## 6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by FSAR Section 17.3. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the On-Site Review Committee (ORC) and the approval of the plant manager.

## 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by FSAR Section 17.3. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

OFFSITE DOSE CALCULATION MANUAL (Continued)

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the On-Site Review Committee (ORC) and the approval of the plant manager.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.15 Specification 6.15 has been deleted from Technical Specifications and has been relocated to the ODCM and PCP, as appropriate.

Page 6-29 has been deleted.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. NPF-63

SHEARON HARRIS NUCLEAR POWER PLANT

UNIT 1

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-400

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

JANUARY 1987

SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NO. 1

ENVIRONMENTAL PROTECTION PLAN
(NONRADIOLOGICAL)

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## 1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of nonradiological environmental values during operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement - Operating Licensing Stage (FES-OL) and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters are regulated by way of the licensee's NPDES permit.

## 2.0 Environmental Protection Issues

In the FES-OL (NUREG-0972) dated October 1983, the staff considered the environmental impacts associated with the operation of the Shearon Harris Nuclear Power Plant, Unit 1. No aquatic/water quality, terrestrial, or noise issues were identified.

## 3.0 Consistency Requirements

### 3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP*. Changes in station design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

Before engaging in additional construction or operational activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measurable nonradiological environmental effects are confined to the on-site areas previously disturbed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

* This provision does not relieve the licensee of the requirements of 10 CFR 50.59.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level; or (3) a matter, not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of the Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

## 3.2 Reporting Related to the NPDES Permit and State Certification

Changes to, or renewals of, the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The licensee shall notify the NRC of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES Permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, and local environmental regulations are not subject to the requirements of Section 3.1.

## 4.0 Environmental Conditions

### 4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events; onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; fish kills; increase in nuisance organisms or conditions (including <u>Corbicula</u>; unanticipated or emergency discharge of waste water or chemical substances; damage to vegetation resulting from cooling tower drift deposition; and station outage or failure of any cooling water intake or service water system components due to biofouling by <u>Corbicula</u>.

No routine monitoring programs are required to implement this condition.

### 4.2 Environmental Monitoring

#### 4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decisions made by the State of North Carolina under the authority of the Clean Water Act for any requirements for aquatic monitoring.

### 4.2.2 Terrestrial Monitoring

Terrestrial monitoring is not required.

### 4.2.3 Noise Monitoring

Noise monitoring is not required.

## 5.0 Administrative Procedures

### 5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the EPP. The audits 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 and audit function and results of the audit activities shall be maintained and made available for inspection.

### 5.2 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the station. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

## 5.3 Changes in Environmental Protection Plan

Requests for changes in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

## 5.4 Plant Reporting Requirements

### 5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The period of the first report shall begin with the date of issuance of the operating license, and the initial report shall be submitted prior to May 1 of the year following issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 (if any) of this EPP for the report period, including a comparison with related preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends toward irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of mitigating action.

The Annual Environmental Operating Report shall also include:

(1) A list of EPP noncompliances and the corrective actions taken to remedy them.

(2) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental question.

(3) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing results shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall: (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact, and plant operating characteristics; (b) describe the probable cause of the event; (c) indicate the action taken to correct the reported event; (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar oecurrences 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

## ANTITRUST CONDITIONS

The licensee, Duke Energy Progress, LLC is subject to the following antitrust conditions:

Commitment No. 1

Licensee recognizes that it is generally in the public interest for electric utilities to interconnect, coordinate reserves, and engage in bulk power supply transactions, in order to increase electric system reliability and reduce the costs of electric power. Bulk power supply arrangements should be such as to provide benefits, on balance, each to licensee and to other participant(s), respectively. The benefits to participants in such arrangements need not be equal and the benefits realized by a small system may be proportionately greater than those realized by a larger system. In implementing the commitments which it makes in the succeeding paragraphs, licensee will act in accordance with the foregoing principles.

Explanatory Note*

(a) Neither licensee nor any other participant shall be obligated to enter into such arrangements (1) if to do so would violate, incapacitate, or limit its ability to perform any other existing contractual arrangement, or (2) to do so would adversely affect its system operations or the reliability of power supply to its customers, or (3) if to do so would jeopardize the licensee's ability to finance or construct on reasonable terms facilities needed to meet its own anticipated system requirements.

Commitment No. 2

Licensee will interconnect with and coordinate reserves by means of the sale and exchange of emergency bulk power with any entity or entities in its service area** engaging in or proposing to engage in electric bulk power supply on terms that will provide for licensee's costs (including a reasonable return) in connection therewith; and allow the other participant(s), as well as licensee, full access on a proportionate basis to the benefits of reserve coordination. ("Proportionate basis" refers to the equalized percentage of reserves concept rather than the largest single-unit concept, unless all participants otherwise agree).

* In order to clarify the commitments, certain explanatory notes have been added.

** The use of the term "service area" as found in this commitment or in any other section of the commitments is intended to describe those areas in North Carolina and South Carolina where licensee provides some class of electric service, but in no way indicates an assignment or allocation of wholesale market areas.

Explanatory Notes

(a) Interconnections will not be limited to low voltages when higher voltages are available from licensee's installed facilities in the area where interconnection is desired, when the proposed arrangement is found to be technically and economically feasible.

(b) Emergency service agreements will not be limited to a fixed amount, but emergency service provided under such agreements will be furnished if and when available and desired where such supply does not impair or threaten to impair service to the supplier's customers due to capacity availability, fuel supply, system reliability or other good cause. Licensee, however, shall not be obligated to provide emergency service to another entity in lieu of such entity's maintaining its own adequate system reserves or fuel supply.

(c) An example of the type of reserve sharing arrangement available to any participant and which would provide "full access on a proportional basis to the benefits of reserve coordination" would be one in which the following conditions would obtain:

(i) The licensee and each participant(s) shall provide to the other emergency power if and when available from its own generation, or through its transmission from the generation of others to the extent it can do so without disrupting or threatening to impair service to its own customers due to capacity availability, fuel supply, system reliability or other good cause.

(ii) The participants to the reserve sharing agreement, including licensee, shall, consistent with licensee's reserve policy as established from time to time by licensee, determine a minimum percentage reserve to be installed and/or purchased by the participants, including licensee, as necessary to maintain in total an adequate and reliable power supply on the interconnected system of licensee and participant(s).

Commitment No. 3

Licensee will purchase from or sell "bulk power" to any other entity in its service area engaging in or proposing to engage in the generation of electric power in bulk at the seller's cost (including a reasonable return) whenever such transactions would serve to reduce the overall costs of new bulk power supply, each, for itself and other participant(s) to the transaction, respectively. ("Costs" refers to costs of bulk power supply determined in accordance with the seller's normal practices, without regard to the purchaser's intended use of the power or the status of the purchaser). This paragraph refers specifically to the opportunity to coordinate in the planning of new generation, transmission and associated facilities. If licensee questions the desirability of a proposed transaction on the ground that it would not reduce its overall bulk power costs, it will make available upon request to the entity proposing the transaction such information as is relevant and reasonably necessary to establish its bulk power costs.

Explanatory Notes

(a) It is not to be considered that this condition requires licensee to purchase or sell bulk power if such purchase or sale is technically infeasible or that the benefits therefrom do not exceed the costs in connection with such purchase or sale.

Commitment No. 4

Licensee will facilitate the exchange of bulk power by transmission over its system between or among two or more entities with which it is interconnected on terms which will fully compensate it for the service performed, to the extent that such arrangements reasonably can be accommodated from a functional and technical standpoint.

Explanatory Notes

(a) This condition applies to entities with which licensee is interconnected in the future as well as to which it is now interconnected.

Commitment No. 5

Licensee will sell power in bulk to any entity in the aforesaid area now engaging in or proposing to engage in the retail distribution of electric power.

Explanatory Notes

(a) This is provided that licensee has such power available for sale after making adequate provision for the capacity, fuel and other requirements of its service area customers.

Commitment No. 6

The implementation of these numbered paragraphs shall be in all respects on reasonable terms and conditions as consistent with the Federal Power Act and all other lawful regulation and authority, and shall be subject to engineering and technical feasibility for licensee's system. Licensee will negotiate (including the execution of a contingent statement of intent) with respect to the foregoing commitments with any entity in its service area engaging in or proposing to engage in bulk power supply transactions, but licensee shall not be required to enter into any final arrangements prior to resolution of any substantial questions as to the lawful authority of an entity to engage in the transactions.

Commitment No. 7

In contracts between licensee and its wholesale customers, licensee will not attempt to restrict such customers from electrically connecting with other sources of power if reasonable written notice to licensee has been made and agreement reached by the parties on such measures or conditions, if any, as may be required for the protection and reliability of both systems.

APPENDIX D

ADDITIONAL CONDITIONS

RENEWED LICENSE NO. NPF-63

Duke Energy Progress, LLC shall comply with the following conditions on the schedule noted below:

| Amendment Number | Additional Conditions | Implementation Date |
|---|---|---|
| ~~174~~, 188 | Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 1 License Amendment No. 174 dated September 17, 2019.<br><br>In addition, Duke Energy is approved to implement 10 CFR 50.69 using the alternative seismic approach for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs as described in Duke Energy letter RA-20-0311, dated January 14, 2021, and as specified in License Amendment No. 188 dated January 19, 2022.<br><br>Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from the alternate seismic approach (referenced above) to a seismic probabilistic risk assessment approach). | Upon implementation of Amendment No. 188. |