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

DOMINION ENERGY NUCLEAR CONNECTICUT, INC., ET AL.[1]

DOCKET NO. 50-423

(MILLSTONE POWER STATION, UNIT NO. 3)

RENEWED FACILITY OPERATING LICENSE

RENEWED LICENSE NO. NPF-49

1. The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. NPF-49 issued on January 31, 1986 has now found that:

   A. The application to renew License NPF-49 filed by Dominion Energy Nuclear Connecticut, Inc. (DENC), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter 1, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Millstone Power Station, Unit No. 3 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-113 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 analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed 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;

---

(1) Dominion Energy Nuclear Connecticut, Inc., et al. (the licensees) consists of Dominion Energy Nuclear Connecticut, Inc., Green Mountain Power Corporation and Massachusetts Municipal Wholesale Electric Company. Dominion Energy Nuclear Energy Connecticut, Inc. is authorized to act as the agent and representative for Green Mountain Power Corporation and Massachusetts Municipal Wholesale Electric Company and has exclusive responsibility and control over the physical operation and maintenance of the facility.

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, see Section 2.D below);

E. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1, (except as exempted from compliance, see Section 2.D below);

F. DENC is technically qualified to engage in the activities authorized by this renewed license in accordance with the Commission's regulations set forth in 10 CFR Chapter 1;

G. The licensees have satisfied theapplicable provisions of 10 CFR Part 140 "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations;

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

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 Renewed Facility Operating License No. NPF-49, 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; and

J. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and70.

2. On the basis of the foregoing findings regarding this facility, Facility Operating License No. NPF-49, issued on January 31, 1986, is superceded by Renewed Facility Operating License No. NPF-49, which is hereby issued to Dominion Energy Nuclear Connecticut, Inc. to read as follows:

A. This renewed operating license applies to the Millstone Power Station (MPS), Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in Waterford Township, New London County, Connecticut on the north shore of Long Island Sound, and is described in the licensees' "Final Safety Analysis Report," as supplemented and amended, and in the licensees' Environmental Report, as supplemented and amended.

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

(1) DENC, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in New London County, Connecticut in accordance with the procedures and limitations set forth in this license; Green Mountain Power Corporation and Massachusetts Municipal Wholesale Electric Company, pursuant to the Act and 10 CFR Part 50, to possess the facility at the designated location in New London County, Connecticut in accordance with the procedures and limitations set forth in this renewed operating license;

(2) DENC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) DENC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(5) DENC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter 1 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

DENC is authorized to operate the facility at reactor core power levels not in excess of 3709 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No. 288 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DENC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) DENC shall not take any action that would cause Dominion Energy, Inc. or its parent companies to void, cancel, or diminish DENC's Commitment to have sufficient funds available to fund an extended plant shutdown as represented in the application for approval of the transfer of the licenses for MPS Unit No. 3.

(4) Immediately after the transfer of interests in MPS Unit No. 3 to DNC*, the amount in the decommissioning trust fund for MPS Unit No. 3 must, with respect to the interest in MPS Unit No. 3, that DNC* would then hold, be at a level no less than the formula amount under 10 CFR 50.75.

(5) The decommissioning trust agreement for MPS Unit No. 3 at the time the transfer of the unit to DNC* is effected and thereafter is subject to the following:

(a) The decommissioning trust agreement must be in a form acceptable to the NRC.

(b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Dominion Energy, Inc. or its affiliates or subsidiaries, successors, or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(c) The decommissioning trust agreement for MPS Unit No. 3 must provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee until the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.

(d) The decommissioning trust agreement must provide that the agreement cannot be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

* On May 12, 2017, the name “Dominion Nuclear Connecticut, Inc.” changed to “Dominion Energy Nuclear Connecticut, Inc.”

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

(6) DENC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for approval of the transfer of the MPS Unit No. 3 license and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order.

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

(8) The FSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. DENC shall complete these activities no later than November 25, 2025, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

(9) All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

(10) <u>Mitigation Strategy License Condition</u>

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

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aide 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

(11) Upon implementation of Amendment No. 243 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 4.7.7.h, in accordance with TS 6.8.4.h.c.(i), the assessment of CRE habitability as required by TS 6.8.4.h.c.(ii), and the measurement of CRE pressure as required by TS 6.8.4.h.d, shall be considered met. Following implementation:

(a) The first performance of SR 4.7.7.h, in accordance with TS 6.8.4.h.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from June 16, 2004, the date of the most recent successful tracer gas test, as identified in the report referenced in the August 31, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, TS 6.8.4.h.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from June 16, 2004, the date of the most recent successful tracer gas test, as identified in the report referenced in the August 31, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, TS 6.8.4.h.d, shall be within 24 months, plus the 180 days allowed by SR 4.0.2, as measured from March 23, 2007, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

D. Exemptions from certain requirements of Appendix J to 10 CFR Part 50 (Section 6.2.6, SSER 4) and from a portion of the requirements of General Design Criterion 4 (Section 3.9.3.1, SSER 4) of Appendix A to 10 CFR Part 50 have previously been granted. See Safety Evaluation Report Supplement 4, November 1985. With these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training, and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, submitted by letter dated October 15, 2004, as supplemented by letter dated May 15, 2006, is entitled: "Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 0" The set contains Safeguards Information protected under 10 CFR 73.21.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved Kewaunee, Millstone, North Anna, and Surry Power Stations Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The CSP was approved by License Amendment No. 251, as supplemented by a change approved by License Amendment No. 265.

F. Deleted.

G. 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.

H. <u>Fire Protection (Section 9.5.1, SER, SSER 2, SSER 4, SSER5)</u>

DENC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SER (NUREG-1031) issued July 1985 and Supplements Nos. 2, 4, and 5 issued September 1985, November 1985, and January 1986, respectively, subject to the following provision:

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

I. This renewed operating license is effective as of its date of issuance and shall expire at midnight on November 25, 2045.

FOR THE NUCLEAR REGULATORY COMMISSION

**/ RA /**

J. E. Dyer, Director

Office of Nuclear Reactor Regulation

Attachments:

1. Appendix A - Technical Specifications

2. Appendix B - Environmental Protection Plan

Date of Issuance: November 28, 2005

NUREG-1176



# Technical Specifications

## Millstone Nuclear Power Station, Unit No. 3

Docket No. 50-423

Appendix "A" to
License No. NPF-49

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

January 1986

LICENSE AUTHORITY FILE COPY
DO NOT REMOVE

PAGES i THROUGH xix HAVE BEEN INTENTIONALLY DELETED

# 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 for valves that are opened under administrative control as permitted by 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 the Containment Leakage Rate Testing Program, and

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

1.8 DELETED

CORE ALTERATIONS

1.9 CORE ALTERATIONS 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.

DOSE EQUIVALENT I-131

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

* In MODE 4, the requirement for an OPERABLE containment isolation valve system is satisfied by use of the containment isolation actuation pushbuttons.

DOSE EQUIVALENT XE-133

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

1.12 DELETED

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.

1.14 DELETED

FREQUENCY NOTATION

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

LEAKAGE

1.16 LEAKAGE shall be:

1.16.1 CONTROLLED LEAKAGE

CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals, and

1.16.2 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

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

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

1.16.3 PRESSURE BOUNDARY LEAKAGE

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

1.16.4 UNIDENTIFIED LEAKAGE

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

MASTER RELAY TEST

1.17 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 continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.18 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.

The term “REAL MEMBER OF THE PUBLIC” means an individual who is exposed to existing dose pathways at one particular location.

OPERABLE - OPERABILITY

1.19 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

1.20 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

1.21 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.

1.22 DELETED

PURGE - PURGING

1.23 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

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

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3650 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 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

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

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

SLAVE RELAY TEST

1.32 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.

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

STAGGERED TEST BASIS

1.34 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

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 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.

1.37 DELETED

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access 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.

VENTING

1.39 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.

1.40 Deleted

1.41 Deleted

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

1.43 Deleted

1.44 Deleted

**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

## 2.1 SAFETY LIMITS

### REACTOR CORE

2.1.1 The combination of THERMAL POWER, Reactor Coolant System highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT; and the following Safety Limits shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.14 for the WRB-2M DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 9°F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the Reactor Core Safety Limit 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 2750 psia.

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia 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 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

THIS PAGE INTENTIONALLY LEFT BLANK

This page intentionally left blank.

## 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 Nominal Trip Setpoint values shown in Table 2.2-1.

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

ACTION:

a. With a Reactor Trip System Instrumentation Channel or Interlock Channel Nominal Trip Setpoint inconsistent with the value shown in the Nominal Trip Setpoint column of Table 2.2-1, adjust the Setpoint consistent with the Nominal Trip Setpoint value.

b. With a Reactor Trip System Instrumentation Channel or Interlock Channel found to be inoperable, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. |
| 2. Power Range, Neutron Flux | | |
| a. High Setpoint | 109% of RTP** | ≤ 109.6% of RTP** |
| b. Low Setpoint | 25% of RTP** | ≤ 25.6% of RTP** |
| 3. Power Range, Neutron Flux, High Positive Rate | 5% of RTP** with a time constant ≥ 2 seconds | ≤ 5.6% of RTP** with a time constant ≥ 2 seconds |
| 4. Deleted | | |
| 5. Intermediate Range, Neutron Flux | 25% of RTP** | ≤ 27.4% of RTP** |
| 6. Source Range, Neutron Flux | 1 X $10^{+5}$cps | ≤ 1.06 x $10^{+5}$ cps |
| 7. Overtemperature ΔT | See Note 1 | See Note 2 |

**RTP = RATED THERMAL POWER

## TABLE 2.2-1 (Continued)
## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 8. Overpower $\Delta T$ | See Note 3 | See Note 4 |
| 9. Pressurizer Pressure-Low | 1900 psia | $\geq$ 1897.6 psia |
| 10. Pressurizer Pressure-High | 2385 psia | $\leq$ 2387.4 psia |
| 11. Pressurizer Water Level-High | 89% of instrument span | $\leq$ 89.3% of instrument span |
| 12. Reactor Coolant Flow-Low | 90% of nominal loop flow | $\geq$ 89.8% of nominal loop flow |
| 13. Steam Generator Water Level Low-Low | 18.1% of narrow range instrument span | $\geq$ 17.8% of narrow range instrument span |
| 14. General Warning Alarm | N.A. | N.A. |
| 15. Low Shaft Speed - Reactor Coolant Pumps | 92.4% of rated speed | $\geq$ 92.2% of rated speed |

**TABLE 2.2-1 (Continued)**
**REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS**

| FUNCTIONAL UNIT | | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|
| 16. | Turbine Trip | | |
| | a. Low Fluid Oil Pressure | 500 psig | ≥ 450 psig |
| | b. Turbine Stop Valve Closure | 1% open | ≥ 1% open |
| 17. | Safety Injection Input from ESF | N.A. | N.A. |
| 18. | Reactor Trip System Interlocks | | |
| | a. Intermediate Range Neutron Flux, P-6 | $1 \times 10^{-10}$ amp | $\geq 9.0 \times 10^{-11}$ amp |
| | b. Low Power Reactor Trips Block, P-7 | | |
| | 1) Power Range Neutron Flux, P-10 input (Note 5) | 11% of RTP** | ≤ 11.6% of RTP** |
| | 2) Turbine Impulse Chamber Pressure, P-13 input | 10% RTP** Turbine Impulse Pressure Equivalent | ≤ 10.6% RTP** Turbine Impulse Pressure Equivalent |
| | c. Power Range Neutron Flux, P-8 | 50.0% of RTP** | ≤ 50.6% of RTP** |

** RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| d. Power Range Neutron Flux, P-9 | 51% of RTP** | $\leq$ 51.6% of RTP** |
| e. Power Range Neutron Flux, P-10 (Note 6) | 9% of RTP** | $\geq$ 8.4% of RTP** |
| 19. Reactor Trip Breakers | N.A. | N.A. |
| 20. Automatic Trip and Interlock Logic | N.A. | N.A. |
| 21. DELETED | | |

**RTP = RATED THERMAL POWER

## TABLE 2.2-1 (Continued)
## TABLE NOTATIONS

NOTE 1:OVERTEMPERATURE $\Delta T$

$$\left(\frac{\Delta T}{\Delta T_0}\right)\frac{(1+\tau_1 s)}{(1+\tau_2 s)} \le K_1 - K_2\frac{(1+\tau_4 s)}{(1+\tau_5 s)}(T-T') + K_3(P-P') - f_1(\Delta I)$$

Where: $\Delta T$ is measured Reactor Coolant System $\Delta T$, °F;

$\Delta T_0$ is loop specific indicated $\Delta T$ at RATED THERMAL POWER, °F;

$\frac{(1+\tau_1 s)}{(1+\tau_2 s)}$ is the function generated by the lead-lag compensator on measured $\Delta T$;

$\tau_1$ and $\tau_2$ are the time constants utilized in the lead-lag compensator for $\Delta T$, $\tau_1 \geq$ [*] sec, $\tau_2 \leq$ [*] sec;

$K_1 \leq$ [*]

$K_2 \geq$ [*]/°F;

$\frac{(1+\tau_4 s)}{(1+\tau_5 s)}$ is the function generated by the lead-lag compensator for $T_{avg}$;

$\tau_4$ and $\tau_5$ are the time constants utilized in the lead-lag compensator for $T_{avg}$, $\tau_4 \geq$ [*] sec, $\tau_5 \leq$ [*] sec;

T is measured Reactor Coolant System average temperature, °F;

T' is loop specific indicated $T_{avg}$ at RATED THERMAL POWER, $\leq$ [*]°F;

$K_3 \geq$ [*]/psi

P is measured pressurizer pressure, psia;

P' is nominal pressurizer pressure, $\geq$ [*] psia;

s is the Laplace transform operator, $sec^{-1}$;

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

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: (Continued)

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

(1) For $q_t - q_b$ between [*]% and [*]%, $f_1$, ($\Delta I$) $\geq$ [*], where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RATED THERMAL POWER;

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

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

NOTE 2: The maximum channel as left trip setpoint shall not exceed its computed trip setpoint by more than the following:

(1) 0.4% $\Delta T$ span for the $\Delta T$ channel

(2) 0.4% $\Delta T$ span for the $T_{avg}$ channel

(3) 0.4% $\Delta T$ span for the pressurizer pressure channel

(4) 0.8% $\Delta T$ span for the f($\Delta I$) channel

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

## TABLE 2.2-1 (Continued)

## TABLE NOTATIONS

NOTE 3: OVERPOWER $\Delta T$

$$\left(\frac{\Delta T}{\Delta T_0}\right)\frac{(1+\tau_1 s)}{(1+\tau_2 s)} \le K_4 - K_6(T - T'')$$

Where: $\Delta T$ is measured Reactor Coolant System $\Delta T$, °F;

$\Delta T_0$ is loop specific indicated $\Delta T$ at RATED THERMAL POWER, °F;

$\frac{(1+\tau_1 s)}{(1+\tau_2 s)}$ is the function generated by the lead-lag compensator on measured $\Delta T$;

$\tau_1$ and $\tau_2$ are the time constants utilized in the lead-lag compensator for $\Delta T$, $\tau_1 \ge$ [*] sec, $\tau_2 \le$ [*] sec;

$K_4 \le$ [*];

T is measured average Reactor Coolant System temperature, °F;

T" is loop specific indicated $T_{avg}$ at RATED THERMAL POWER, $\le$ [*]°F;

$K_6 \ge$ [*]/°F when T > T" and $K_6 \le$ [*]/°F when T ≤ T";

s is the Laplace transform operator, $sec^{-1}$;

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

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

## 3/4.0 APPLICABILITY

### LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in Specification 3.0.5.

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, except as provided in Specification 3.0.5. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

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

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

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

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

3.0.5 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 Specifications 3.0.1 and 3.0.2 for the system returned to service under administrative controls to perform the testing required to demonstrate OPERABILITY.

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, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified surveillance interval shall be failure to meet the Limiting Condition for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

## LIMITING CONDITION FOR OPERATION

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the 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 Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

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

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

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

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

4.0.5 Surveillance Requirements for inservice testing of ASME Code Class 1, 2, and 3 components shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) as follows:

a. Deleted.

b. Surveillance intervals specified in the ASME OM Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

| ASME OM Code and applicable Addenda terminology for inservice testing activities | Required frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies and to other normal and accelerated frequencies specified as 2 years or less in the inservice testing program for performing inservice testing activities;

d. Performance of the above inservice testing activities shall be in addition to other specified Surveillance Requirements; and

e. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

f. The provisions of Specification 4.0.3 are applicable to inservice testing activities.

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### 3/4.1.1.1 BORATION CONTROL

### SHUTDOWN MARGIN - MODES 1 AND 2

### LIMITING CONDITION FOR OPERATION

3.1.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the SHUTDOWN MARGIN not within the limits specified in the COLR, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 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 within the limits 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 immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or 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. When in MODE 2 with $K_{eff}$ less than 1, 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;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.2, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

---

* See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% $\Delta k/k$ at the frequency specified in the Surveillance Frequency Control Program. This comparison shall consider at least 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.

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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 3, 4 AND 5 LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.1.1.1.2 The SHUTDOWN MARGIN shall be within the limits 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 33 gpm of a solution containing greater than or equal to 6600 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be within the limits 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 immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or 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.

4.1.1.1.2.2 Valve 3CHS*V305 shall be verified closed and locked at the frequency specified in the Surveillance Frequency Control Program.

---

* Additional SHUTDOWN MARGIN requirements, if required, are given in Specification 3.3.5.

THIS PAGE INTENTIONALLY LEFT BLANK

This page intentionally left blank.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to

a. the limits specified in the CORE OPERATING LIMITS REPORT (COLR) for MODE 5 with RCS loops not filled* or

b. the limits specified in the COLR for MODE 5 with RCS loops filled* with the chemical and volume control system (CVCS) aligned to preclude reactor coolant system boron concentration reduction.

APPLICABILITY: MODE 5 LOOPS NOT FILLED

ACTION:

a. With the SHUTDOWN MARGIN less than the above, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

b. With the CVCS dilution flow paths not closed and secured in position in accordance with Specification 3.1.1.2(b), immediately close and secure the paths or meet the limits specified in the COLR for MODE 5 with RCS loops not filled.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be within the limits 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 immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or 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,

---

* Additional SHUTDOWN MARGIN requirements, if required, are given in Specification 3.3.5.

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.2.2 At the frequency specified in the Surveillance Frequency Control Program the following valves shall be verified closed and locked. The valves may be opened on an intermittent basis under administrative controls except as noted.

| Valve Number | | Valve Function | Valve Position |
|---|---|---|---|
| 1. | V304(Z-) | Primary Grade Water to CVCS | Closed |
| 2. | V120(Z-) | Moderating Hx Outlet | Closed |
| 3. | V147(Z-) | BTRS Outlet | Closed |
| 4. | V797(Z-) | Failed Fuel Monitoring Flushing | Closed |
| 5. | V100(Z-) | Resin Sluice, CVCS Cation Bed Demineralizer | Closed |
| 6. | V571(Z-) | Resin Sluice, CVCS Cation Bed Demineralizer | Closed |
| 7. | V111(Z-) | Resin Sluice, CVCS Cation Bed Demineralizer | Closed |
| 8. | V112(Z-) | Resin Sluice, CVCS Cation Bed Demineralizer | Closed |
| 9. | V98(Z-)/V99(Z-) | Resin Sluice, CVCS Mixed Bed Demineralizer | Closed |
| 10. | V569(Z-)/V570(Z-) | Resin Sluice, CVCS Mixed Bed Demineralizer | Closed |
| 11. | V107(Z-)/V109(Z-) | Resin Sluice, CVCS Mixed Bed Demineralizer | Closed |
| 12. | V108(Z-)/V110(Z-) | Resin Sluice, CVCS Mixed Bed Demineralizer | Closed |
| 13. | V305(Z-)* | Primary Grade Water to Charging Pumps | Closed |

* This valve may not be opened under administrative controls.

THIS PAGE INTENTIONALLY LEFT BLANK

## MODERATOR TEMPERATURE COEFFICIENT

## LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORTS (COLR). The maximum upper limit shall be less positive than $+0.5 \times 10^{-4}$ Δk/k/°F for all the rods withdrawn, beginning of cycle life (BOL), condition for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 Δk/k/°F at 100% RATED THERMAL POWER.

APPLICABILITY: BOL - MODES 1 and 2* only**.
End of Cycle life (EOL) Limit - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the BOL limit of Specification 3.1.1.3 above, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the above limits 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;

   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; and

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

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

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

**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 BOL limit of Specification 3.1.1.3, above, 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 (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

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.

This Page Intentionally Left Blank

This Page Intentionally Left Blank

This Page Intentionally Left Blank

This Page Intentionally Left Blank

This Page Intentionally Left Blank

This Page Intentionally Left Blank

This Page Intentionally Left Blank

## 3/4.1.3 MOVABLE CONTROL ASSEMBLIES

### GROUP HEIGHT

### LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

b. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or 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 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;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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. 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.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ±12 steps of the inoperable rods 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, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length 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 full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at the frequency specified in the Surveillance Frequency Control Program.

## TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly 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)

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 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:

   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. Verify the position of the nonindicating rod(s) indirectly using the movable incore detectors within 8 hours and once per 31 days thereafter, and within 8 hours after discovery of each unintended rod movement, and within 8 hours after each movement of the nonindicating rod(s) greater than 12 steps, and prior to THERMAL POWER exceeding 50% RATED THERMAL POWER, and within 8 hours after reaching RATED THERMAL POWER or

   3. 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:

   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

4.1.3.2.1 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.

4.1.3.2.2 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.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (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 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 65% of RATED THERMAL POWER with the reactor coolant stop valves in the nonoperating loop closed.

SURVEILLANCE REQUIREMENTS

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

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

b. Deleted

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

SHUTDOWN BANK INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 Each shutdown bank shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown bank inserted beyond the insertion limits specified in the COLR except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the bank to within the limit specified in the COLR within 1 hour, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown bank shall be determined to be within the insertion limits 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.

* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

CONTROL BANK 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 limits specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits 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 limits 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.

VENTING

1.39 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.

SPENT FUEL POOL STORAGE PATTERNS:

STORAGE PATTERN

1.40 STORAGE PATTERN refers to the blocked location in a Region 1 fuel storage rack and all adjacent and diagonal Region 1 (or Region 2) cell locations surrounding the blocked location. The blocked location is for criticality control.

3-OUT-OF-4 AND 4-OUT-OF-4

1.41 Region 1 spent fuel racks can store fuel in either of 2 ways:

(a) Areas of the Region 1 spent fuel racks with fuel allowed in every storage location are referred to as the 4-OUT-OF-4 Region 1 storage area.

(b) Areas of the Region 1 spent fuel racks which contain a cell blocking device in every 4th location for criticality control, are referred to as the 3-OUT-OF-4 Region 1 storage area. A STORAGE PATTERN is a subset of the 3-OUT-OF-4 Region 1 storage area.

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

1.43 Deleted

1.44 Deleted

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AXIAL FLUX DIFFERENCE

### LIMITING CONDITION FOR OPERATION

3.2.1.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. The limits specified in the CORE OPERATING LIMITS REPORT (COLR)

b. Deleted

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

ACTION:

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

   1. Either restore the indicated AFD to within the COLR specified limits 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. Deleted

c. 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

4.2.1.1.1 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:

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.1.2 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.

4.2.1.1.3 Deleted

4.2.1.1.4 Deleted

This page intentionally left blank.

This page intentionally left blank.

POWER DISTRIBUTION LIMITS

## 3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

### LIMITING CONDITION FOR OPERATION

3.2.2.1 $F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. With Specification 4.2.2.1.2.b not being satisfied:

(1) Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower DT Trip setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit, and

(2) Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by item (1) above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

b. With Specification 4.2.2.1.2.c not being satisfied, all of the following ACTIONS shall be taken:

(1) a. Within 4 hours, control the AFD to within the new reduced AFD limits specified in the COLR that restores $F_Q(Z)$ to within its limits, and

b. Reduce the THERMAL POWER by the amount specified in the COLR that restores $F_Q(Z)$ to within its limits within 4 hours, and

c. Reduce the Power Range Neutron Flux - High Trip Setpoints by $\geq$1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

d. Reduce the Overpower $\Delta T$ Trip Setpoints by $\geq$1% for each 1% that the THERMAL POWER level is reduced within 72 hours, and

e. Within 8 hours, reset the AFD Alarm Setpoints to the modified limits, and

f. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION b(1)b above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

(2) Deleted

c. Deleted

THIS PAGE INTENTIONALLY LEFT BLANK

LIMITING CONDITION FOR OPERATION (Continued)

SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

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

b. Evaluate the computed heat flux hot channel factor by performing both of the following:

(1) Determine the computed heat flux hot channel Factor, $F_Q^M(Z)$ by increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties, and

(2) Verify that $F_Q^M(Z)$ satisfies the requirements of Specification 3.2.2.1 for all core plane regions, i.e., 0-100% inclusive.

c. Verify $F_Q^M(Z)$ satisfies the non-equilibrium limits specified in the COLR.

d. Measuring $F_Q^M(Z)$ 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 $F_Q(Z)$ was last determined,*** or

(2) At the frequency specified in the Surveillance Frequency Control Program, whichever occurs first.

e. Compliance with the non-equilibrium limits shall be conservatively accounted for during intervals between $F_Q^M(Z)$ measurements by performing either of the following:

(1) Increase $F_Q^M(Z)$ by an appropriate factor specified in the COLR and verify that this value satisfies Specification 4.2.2.1.2.c, or

(2) Verify $F_Q^M(Z)$ satisfies its limits at least once per 7 Effective Full Power Days.

f. The limits specified in Specifications 4.2.2.1.2c and 4.2.2.1.2e above are not applicable in the core plane regions defined in the Bases.

4.2.2.1.3 Deleted

4.2.2.1.4 Deleted

4.2.2.1.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specifications 4.2.2.1.2, an overall measured $F_Q(Z)$ 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.

***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 power distribution map outlined.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

This page intentionally left blank.

This page intentionally left blank.

This page intentionally left blank.

This page intentionally left blank.

This page intentionally left blank.

This page intentionally left blank.

This page intentionally left blank.

## 3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

### LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate ≥ 363,200 gpm and greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR), and

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

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}},$

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER in the COLR,

4) $PF_{\Delta H}$ = The power factor multiplier for $F^N_{\Delta H}$ provided in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate to within the limits specified above and in the COLR and $F^N_{\Delta H}$ to within the above limit, or

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

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that the RCS total flow rate is restored to within the limits specified above and in the COLR and $F_{\Delta H}^{N}$ is restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}^{N}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 $F_{\Delta H}^{N}$ shall be determined to be within the acceptable range:

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

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

4.2.3.1.3 The RCS total flow rate shall be determined to be within the acceptable range by:

a. Verifying by precision heat balance that the RCS total flow rate is ≥ 363,200 gpm and greater than or equal to the limit specified in the COLR within 7 days after reaching 90% of RATED THERMAL POWER after each fuel loading, and

SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying that the RCS total flow rate is ≥ 363,200 gpm and greater than or equal to the limit specified in the COLR at the frequency specified in the Surveillance Frequency Control Program.

4.2.3.1.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program.

4.2.3.1.5 DELETED.

4.2.3.1.6 DELETED.

This page intentionally left blank.

This page intentionally left blank.

## 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 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, 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.

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 limits specified in the CORE OPERATING LIMITS REPORT (COLR):

a. Reactor Coolant System $T_{avg}$, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the above DNB-related parameters shall be verified to be within the limits specified in the COLR at the frequency specified in the Surveillance Frequency Control Program.

THIS PAGE INTENTIONALLY LEFT BLANK

## 3/4.3 INSTRUMENTATION

### 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 channel 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 at the frequency specified in the Surveillance Frequency Control Program. Neutron detectors and speed sensors are exempt from response time verification. Each verification shall include at least one train and one channel (to include input relays to both trains) per function.

## 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* | 11 |
| 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. Deleted | | | | | |
| 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* | 11 |
| 7. Overtemperature ΔT | 4 | 2 | 3 | 1, 2 | 6A |
| 8. Overpower ΔT | 4 | 2 | 3 | 1, 2 | 6A |
| 9. Pressurizer Pressure--Low | 4 | 2 | 3 | 1** | 6A (1) |
| 10. Pressurizer Pressure--High | 4 | 2 | 3 | 1, 2 | 6A (1) |
| 11. Pressurizer Water Level--High | 3 | 2 | 2 | 1** | 6A |

**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 | 2/loop | 1 | 6 |
| | b. | Two Loops (Above P-7 and below P-8) | 3/loop | 2/loop in two operating loops | 2/loop | 1 | 6 |
| 13. | Steam Generator Water Level--Low-Low | | 4/stm. gen. | 2/stm. gen. | 3/stm. gen. | 1, 2 | 6A (1) |
| 14. | Low Shaft Speed--Reactor Coolant Pumps | | 4-1/pump | 2 | 3 | 1** | 6A |
| 15. | Turbine Trip | | | | | | |
| | a. | Low Fluid Oil Pressure | 3 | 2 | 2 | 1*** | 12 |
| | b. | Turbine Stop Valve Closure | 4 | 4 | 4 | 1*** | 6A |
| 16. | Deleted | | | | | | |
| 17. | Reactor Trip System Interlocks | | | | | | |
| | a. | Intermediate Range Neutron Flux, P-6 | 2 | 1 | 2 | 2## | 8 |
| | b. | Low Power Reactor Trips Block, P-7 | | | | | |
| | | Power Range Neutron Flux, P-10 Input or | 4 | 2 | 3 | 1 | 8 |
| | | Turbine Impulse Chamber Pressure, P-13 Input | 2 | 1 | 2 | 1 | 8 |

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 17. Reactor Trip System Interlocks (Continued) | | | | | |
| c. Power Range Neutron Flux, P-8 | 4 | 2 | 3 | 1 | 8 |
| d. Power Range Neutron Flux, P-9 | 4 | 2 | 3 | 1 | 8 |
| e. Power Range Neutron Flux, P-10 | 4 | 2 | 3 | 1,2 | 8 |
| 18. Reactor Trip Breakers(2) | 2 | 1 | 2 | 1, 2 | 10, 13 |
| | 2 | 1 | 2 | 3*, 4*, 5* | 11 |
| 19. Automatic Trip and Interlock Logic | 2 | 1 | 2 | 1, 2 | 13A |
| | 2 | 1 | 2 | 3*, 4*, 5* | 11 |
| 20. DELETED | | | | | |
| 21. DELETED | | | | | |

## TABLE 3.3-1 (Continued)

## TABLE NOTATIONS

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

** Above the P-7 (At Power) Setpoint.

*** Above the P-9 (Reactor Trip/Turbine Trip Interlock) Setpoint.

## 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 and ACTION statements for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

(2) Including any reactor trip bypass breakers that are racked in and closed for bypassing a reactor trip breaker.

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 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 72 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1, 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 78 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 additions.*

ACTION 5 - (Not used)

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, 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.

---

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

## TABLE 3.3-1 (Continued)

## ACTION STATEMENTS (Continued)

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

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

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - (Not used)

ACTION 8 - 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 9- (Not used)

ACTION 10- With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY 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 11- 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 12- With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

b. When the Minimum Channels OPERABLE requirement is met, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of the Turbine Control Valves.

ACTION 13- With one of the diverse trip features (undervoltage or shunt trip attachments) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. 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 13A- With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 24 hours or be in at least HOT STANDBY 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.

This page is intentionally left blank

This page is intentionally left blank

**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(14) | 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, 5) | S/U(1) | N.A. | N.A. | 1***, 2 |
| 3. Power Range, Neutron Flux, High Positive Rate | N.A. | SFCP(4, 5) | SFCP | N.A. | N.A. | 1, 2 |
| 4. Deleted | | | | | | |
| 5. Intermediate Range | 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(9) | N.A. | N.A. | 2**, 3*, 4*, 5* |
| 7. Overtemperature ΔT | SFCP | SFCP | 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(18) | N.A. | N.A. | 1***** |
| 10. Pressurizer Pressure--High | SFCP | SFCP | SFCP(18) | N.A. | N.A. | 1, 2 |
| 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 |

**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 |
|---|---|---|---|---|---|---|
| 13. Steam Generator Water Level--Low-Low | SFCP | SFCP | SFCP(18) | N.A. | N.A. | 1, 2 |
| 14. Low Shaft Speed - Reactor Coolant Pumps | N.A. | SFCP(13) | SFCP | N.A. | N.A. | 1 |
| 15. Turbine Trip | | | | | | |
| a. Low Fluid Oil Pressure | N.A. | SFCP | N.A. | S/U(1, 10)**** | N.A. | 1 |
| b. Turbine Stop Valve Closure | N.A. | SFCP | N.A. | S/U(1, 10)**** | N.A. | 1 |
| 16. Deleted | | | | | | |
| 17. Reactor Trip System Interlocks | | | | | | |
| a. Intermediate Range Neutron Flux, P-6 | N.A. | SFCP(4) | SFCP | N.A. | N.A. | 2** |
| 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-9 | N.A. | SFCP(4) | SFCP | N.A. | N.A. | 1 |
| e. Power Range Neutron Flux, P-10 | N.A. | SFCP(4) | SFCP | N.A. | N.A. | 1, 2 |
| f. Turbine Impulse Chamber Pressure, P-13 | N.A. | SFCP | SFCP | N.A. | N.A. | 1 |

## 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 |
|---|---|---|---|---|---|---|
| 18. Reactor Trip Breaker | N.A. | N.A. | N.A. | SFCP(7,11) | N.A. | 1, 2, 3*, 4*, 5* |
| 19. Automatic Trip and Interlock Logic | N.A. | N.A. | N.A. | N.A. | SFCP(7) | 1, 2, 3*, 4*, 5* |
| 20. DELETED | | | | | | |
| 21. Reactor Trip Bypass Breaker | N.A. | N.A. | N.A. | SFCP(7,15)<br>SFCP(16) | N.A. | 1, 2, 3*, 4*, 5* |
| 22. DELETED | | | | | | |

## 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.

**** Above the P-9 (Reactor Trip/Turbine Interlock) Setpoint.

***** Above the P-7 (At Power) 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 Source Range, 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) (Not used)

(9) 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.

## TABLE 4.3-1 (Continued)

## TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) (not used)

(13) Reactor Coolant Pump Shaft Speed Sensor may be excluded from CHANNEL CALIBRATION.

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(15) Local manual shunt trip prior to placing breaker in service.

(16) Automatic undervoltage trip.

(17) (not used).

(18) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 should be reviewed for applicability.

## 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 Nominal Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation Channel or Interlock Channel Nominal Trip Setpoint inconsistent with the value shown in the Nominal Trip Setpoint column of Table 3.3-4, adjust the Setpoint consistent with the Nominal Trip Setpoint value.

b. With an ESFAS Instrumentation Channel or Interlock Channel found to be inoperable, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status.

## 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 the limit at the frequency specified in the Surveillance Frequency Control Program. Each verification shall include at least one train and one channel (to include input relays to both trains) per function.

* The provisions of Specification 4.0.4 are not applicable for response time verification of steam line isolation for entry into MODE 4 and MODE 3 and turbine driven auxiliary feedwater pump for entry into MODE 3.

## 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 Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water). | | | | | |
| a. Manual Initiation | 2 | 1 | 2 | 1, 2, 3, 4 | 19 |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14A |
| c. Containment Pressure--High-1 | 3 | 2 | 2 | 1, 2, 3 | 20A |
| d. Pressurizer Pressure--Low | 4 | 2 | 3 | 1, 2, 3# | 20A |
| e. Steam Line Pressure--Low | 3/steam line in each operating loop | 2/steam line in any operating loop | 2/steam line in each operating loop | 1, 2, 3# | 20A |
| 2. Containment Spray (CDA) | | | | | |
| a. Manual Initiation | 2 | 1 with 2 coincident switches | 2 | 1, 2, 3, 4 | 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 (CDA) (Continued) | | | | | |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14A |
| c. Containment Pressure--High-3 | 4 | 2 | 3 | 1, 2, 3, 4 | 17 |
| 3. Containment Isolation | | | | | |
| a. Phase "A" Isolation | | | | | |
| 1) Manual Initiation | 2 | 1 | 2 | 1, 2, 3, 4 | 19 |
| 2) Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14A |
| 3) Safety Injection | See Item 1. above for all Safety Injection initiating functions and requirements. | | | | |
| b. Phase "B" Isolation | | | | | |
| 1) Manual Initiation | 2 | 1 with 2 coincident switches | 2 | 1, 2, 3, 4 | 19 |
| 2) Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14A |

## 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) | | | | | |
| 3. Containment Pressure--High-3 | 4 | 2 | 3 | 1, 2, 3, 4 | 17 |
| c. DELETED | | | | | |
| 4. Steam Line Isolation | | | | | |
| a. Manual Initiation | | | | | |
| 1. Individual | 1/steam line | 1/steam line | 1/operating steam line | 1, 2, 3, 4 | 24 |
| 2. System | 2 | 1 | 2 | 1, 2, 3, 4 | 23 |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 22 |
| c. Containment Pressure--High-2 | 3 | 2 | 2 | 1, 2, 3, 4 | 20A |
| d. Steam Line Pressure--Low | 3/steam line in each operating loop | 2/steam line in any operating loop | 2/steam line in each operating loop | 1, 2, 3# | 20A |
| e. Steam Line Pressure - Negative Rate--High | 3/steam line in each operating loop | 2/steam line in any operating loop | 2/steam line in each operating loop | 3**** | 20A |

**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 |
|---|---|---|---|---|---|
| 5. Turbine Trip and Feedwater Isolation | | | | | |
| a. Automatic Actuation Logic and Actuaion Relays | 2 | 1 | 2 | 1, 2 | 25 |
| b. Steam Generator Water Level--High-High (P-14) | 4/stm. gen. in each operating loop | 2/stm. gen. in any operating loop | 3/stm. gen. in each operating loop | 1, 2, 3 | 20A, 21 |
| c. Safety Injection Actuation Logic | 2 | 1 | 2 | 1, 2 | 22 |
| d. $T_{ave}$ Low Coincident with P-4 | 1 $T_{ave}$/loop | 1 $T_{ave}$ in any two loops | 1 $T_{ave}$ in any three loops | 1, 2 | 20 |

**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 | 2 | 1 | 2 | 1, 2, 3 | 23 |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3 | 22 |
| c. Stm. Gen. Water Level--Low-Low | | | | | |
| 1) Start Motor-Driven Pumps | 4/stm. gen. | 2/stm. gen. in any operating stm. gen. | 3/stm. gen. in each operating stm. gen. | 1, 2, 3 | 20A |
| 2) Start Turbine-Driven Pump | 4/stm. gen. | 2/stm. gen. in any 2 operating stm. gen. | 3/stm. gen. in each operating stm. gen. | 1, 2, 3 | 20A |
| 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 | 2 | 1 | 2 | 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 |
|---|---|---|---|---|---|
| 6. Auxiliary Feedwater (Continued) | | | | | |
| f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps | See Item 2. above for all CDA functions and requirements. | | | | |
| 7. Control Building Isolation | | | | | |
| a. Manual Actuation | 2 | 1 | 2 | * | 19 |
| b. Manual Safety Injection Actuation | 2 | 1 | 2 | 1, 2, 3, 4 | 19 |
| c. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 14 |
| d. Containment Pressure--High-1 | 3 | 2 | 2 | 1, 2, 3 | 16 |
| e. Control Building Inlet Ventilation Radiation | 2/intake | 1 | 2/intake | * | 18 |
| 8. Loss of Power | | | | | |
| a. 4 kV Bus Under-voltage-Loss of Voltage | 4/bus | 2/bus | 3/bus | 1, 2, 3, 4 | 27 |
| b. 4 kV Bus Undervoltage-Grid Degraded Voltage | 4/bus | 2/bus | 3/bus | 1, 2, 3, 4 | 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 |
|---|---|---|---|---|---|
| 9. Engineering Safety Features Actuation System Interlocks | | | | | |
| a. Pressurizer Pressure, P-11 | 3 | 2 | 2 | 1, 2, 3 | 21 |
| b. Low-Low $T_{avg}$, P-12 | 4 | 2 | 3 | 1, 2, 3 | 21 |
| c. Reactor Trip, P-4 | 2 | 2 | 2 | 1, 2, 3 | 23 |
| 10. Emergency Generator Load Sequencer | 2 | 1 | 2 | 1, 2, 3, 4 | 15 |
| 11. Cold Leg Injection Permissive, P-19 | 4 | 2 | 3 | 1, 2, 3 | 20 |

## TABLE 3.3-3 (Continued)

## TABLE NOTATIONS

\# The Steamline Isolation Logic and Safety Injection Logic for this trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

* MODES 1, 2, 3, and 4.
During movement of recently irradiated fuel assemblies.

**** 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.

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 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 14A - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY 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 Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; 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 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 17 - 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 72 hours and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1.

**TABLE 3.3-3 (Continued)**

ACTION STATEMENTS

ACTION 18 - a. With one Control Building Inlet Ventilation Radiation Monitor channel inoperable, either restore the inoperable channel to OPERABLE status within 7 days or place the associated train of Control Room Emergency Ventilation System in the emergency mode of operation.$^{+}$ Otherwise, immediately suspend movement of recently irradiated fuel assemblies, if applicable, and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Control Building Inlet Ventilation Radiation Monitor channels inoperable, immediately place one train of Control Room Emergency Ventilation System in the emergency mode of operation, declare one Control Room Emergency Ventilation System train inoperable, and comply with the ACTION requirements of Technical Specification 3.7.7. Otherwise, immediately suspend movement of recently irradiated fuel assemblies, if applicable, and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

$^{+}$Operation of the non-affected Control Room Emergency Ventilation System train, instead of the affected Control Room Emergency Ventilation System train, is permitted to perform required Technical Specifications 3.3.2 and 3.7.7 surveillance testing.

ACTION 19 - 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 be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 20 - 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 4.3.2.1.

ACTION 20A - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 21 - 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 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY 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 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 24 - 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 declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY 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.

ACTION 26 - DELETED

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 27 - a. 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:

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

2. 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.

b. With the number of OPERABLE channels one less than the Minimum Channels required OPERABLE:

1. Place one channel in bypass and other channel in trip condition within one hour and restore one channel to OPERABLE status in 48 hours,

OR

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

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water) | | |
| a. Manual Initiation | N.A. | N.A. |
| b. Automatic Actuation Logic | N.A. | N.A. |
| c. Containment Pressure--High 1 | 17.7 psia | ≤ 17.9 psia |
| d. Pressurizer Pressure--Low | | |
| 1) Channels I and II | 1892 psia | ≥ 1889.6 psia |
| 2) Channel III and IV | 1892 psia | ≥ 1889.6 psia |
| e. Steam Line Pressure--Low | 658.6 psig* | ≥ 654.7 psig* |
| 2. Containment Spray (CDA) | | |
| a. Manual Initiation | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. |
| c. Containment Pressure--High-3 | 22.7 psia | ≤ 22.9 psia |
| 3. Containment Isolation | | |
| a. Phase "A" Isolation | | |
| 1) Manual Initiation | N.A. | N.A. |

**TABLE 3.3-4**
**ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS**

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 3. Containment Isolation (Continued) | | |
| 2. Automatic Actuation Logic and Actuation Relays | 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 Initiation | N.A. | N.A. |
| 2. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. |
| 3. Containment Pressure--High-3 | 22.7 psia | ≤ 22.9 psia |
| c. DELETED | | |
| 4. Steam Line Isolation | | |
| a. Manual Initiation | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. |
| c. Containment Pressure--High-2 | 17.7 psia | ≤ 17.9 psia |
| d. Steam Line Pressure--Low | 658.6 psig* | ≥ 654.7 psig* |
| e. Steam Line Pressure - Negative Rate--High | 100 psi/s** | ≤ 103.9 psi/s** |

## TABLE 3.3-4

## ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 5. Turbine Trip and Feedwater Isolation | | |
| a. Automatic Actuation Logic Actuation Relays | N.A. | N.A. |
| b. Steam Generator Water Level--High-High (P-14) | 80.5% of narrow range instrument span. | ≤ 80.8% of narrow range instrument span. |
| c. Safety Injection Actuation Logic | See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values. | |
| d. $T_{ave}$ Low Coincident with Reactor Trip (P-4) | 564°F | ≥ 563.6°F |
| 6. Auxiliary Feedwater | | |
| a. Manual Initiation | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. |
| c. Steam Generator Water Level--Low-Low | | |
| 1) Start Motor-Driven Pumps | 18.1% of narrow range instrument span. | ≥ 17.8% of narrow range instrument span. |

TABLE [illegible]-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 6. Auxiliary Feedwater (Continued) | | |
| 2) Start Turbine-Driven Pumps | 18.1% of narrow range instrument span. | ≥ 17.8% of narrow range instrument span. |
| d. Safety Injection | See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values. | |
| e. Loss-of-Offsite Power Start Motor-Driven Pumps | 2800V | ≥ 2720V |
| f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps | See Item 2. above for all CDA Trip Setpoints and Allowable Values. | |
| 7. Control Building Isolation | | |
| a. Manual Actuation | N.A | N.A. |
| b. Manual Safety Injection Actuation | N.A | N.A. |
| c. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. |
| d. Containment Pressure--High 1 | 17.7 psia | ≤ 17.9 psia |
| e. Control Building Inlet Ventilation Radiation | $\leq 1.5 \times 10^{-5} \mu ci/cc$ | $\leq 1.5 \times 10^{-5} \mu ci/cc$ |

**TABLE 3.3-4 (Continued)**
**ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS**

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 8. Loss of Power | | |
| a. 4 kV Bus Undervoltage (Loss of Voltage) | 2800 volts with a ≤ 2 second time delay. | ≥ 2720 volts with a ≤ 2 second time delay. |
| b. 4 kV Bus Undervoltage (Grid Degraded Voltage) | 3730 volts with a ≤ 8 second time delay with ESF actuation or ≤ 300 second time delay without ESF actuation. | ≥ 3706 volts with a ≤ 8 second time delay with ESF actuation or ≤ 300 second time delay without ESF actuation. |
| 9. Engineered Safety Features Actuation System Interlocks | | |
| a. Pressurizer Pressure, P-11 | 1999.7 psia | ≤ 2002.1 psia |
| b. Low-Low $T_{avg}$, P-12 | 553°F | ≥ 552.6°F |
| c. Reactor Trip, P-4 | N.A. | N.A. |
| 10. Emergency Generator Load Sequencer | N.A. | N.A. |
| 11. Cold Leg Injection Permissive, P-19 | 1900 psia | ≥ 1897.6 psia |

## 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 greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

**TABLE 4.3-2**
**ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION**
**SURVEILLANCE REQUIREMENTS**

| 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 Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water) | | | | | | | | |
| 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(4) | 1, 2, 3, 4 |
| c. Containment Pressure-High-1 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| 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 |
| 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(4) | 1, 2, 3, 4 |
| c. Containment Pressure-High-3 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3, 4 |

**TABLE 4.3-2 (Continued)**
**ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION**
**SURVEILLANCE REQUIREMENTS**

| 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 | | | | | | | | |
| 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(4) | 1, 2, 3, 4 |
| 3. Safety Injection | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |
| b. Phase "B" 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(4) | 1, 2, 3, 4 |
| 3. Containment Pressure-High-3 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| c. DELETED | | | | | | | | |
| 4. Steam Line Isolation | | | | | | | | |
| a. Manual Initiation | | | | | | | | |
| 1. Individual | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| 2. System | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3, 4 |

**TABLE 4.3-2 (Continued)**
**ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION**
**SURVEILLANCE REQUIREMENTS**

| 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. Steam Line Isolation (Continued) | | | | | | | | |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP(4) | 1, 2, 3, 4 |
| c. Containment Pressure-High-2 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| d. Steam Line Pressure-Low | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| e. Steam Line Pressure-Negative Rate-High | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 3 |
| 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(4) | 1, 2 |
| b. Steam Generator Water Level-High-High | SFCP | SFCP | SFCP | N.A. | SFCP(1) | SFCP(1) | SFCP(4) | 1, 2, 3 |
| c. Safety Injection Actuation Logic | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2 |
| d. $T_{ave}$ Low Coincident with Reactor Trip (P-4) | N.A. | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2 |

**TABLE 4.3-2 (Continued)**
**ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION**
**SURVEILLANCE REQUIREMENTS**

| 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 | | | | | | | | |
| 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(4) | 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 | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |
| e. Loss-of-Offsite Power | See Item 8. below for all Loss of Power Surveillance. | | | | | | | |
| f. Containment Depressurization Actuation (CDA) | See Item 2. above for all CDA Surveillance Requirements. | | | | | | | |
| 7. Control Building Isolation | | | | | | | | |
| a. Manual Actuation | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | * |
| b. Manual Safety Injection Actuation | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| c. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | SFCP(1) | SFCP(1) | SFCP(4) | 1, 2, 3, 4 |
| d. Containment Pressure--High-1 | 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**

| 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 |
|---|---|---|---|---|---|---|---|---|
| 7. Control Building Isolation (Continued) | | | | | | | | |
| e. Control Building Inlet Ventilation Radiation | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | * |
| 8. Loss of Power | | | | | | | | |
| a. 4 kV Bus Undervoltage (Loss of Voltage) | N.A. | SFCP | N.A. | SFCP(3) | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| b. 4 kV Bus Undervoltage (Grid Degraded Voltage) | N.A. | SFCP | N.A. | SFCP(3) | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| 9. Engineered Safety Features Actuation System Interlocks | | | | | | | | |
| a. Pressurizer Pressure, P-11 | N.A. | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| b. Low-Low $T_{avg}$, P-12 | N.A. | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| c. Reactor Trip, P-4 | N.A. | N.A. | N.A. | SFCP | N.A. | N.A. | N.A. | 1, 2, 3 |
| 10. Emergency Generator Load Sequencer | N.A. | N.A. | N.A. | N.A. | SFCP(1,2) | N.A. | N.A. | 1, 2, 3, 4 |
| 11. Cold Leg Injection Permissive, P-19 | SFCP | SFCP | SFCP | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |

## TABLE 4.3-2 (Continued)

## TABLE NOTATION

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

2. This surveillance may be performed continuously by the emergency generator load sequencer auto test system as long as the EGLS auto test system is demonstrated OPERABLE by the performance of an ACTUATION LOGIC TEST at the frequency specified in the Surveillance Frequency Control Program.

3. At the frequency specified in the Surveillance Frequency Control Program, a loss of voltage condition will be initiated at each undervoltage monitoring relay to verify individual relay operation. Setpoint verification and actuation of the associated logic and alarm relays will be performed as part of the CHANNEL CALIBRATION.

4. For Engineered Safety Features Actuation System functional units with only Potter & Brumfield MDR series relays used in a clean, environmentally controlled cabinet, as discussed in Westinghouse Owners Group Report WCAP- 13900, the surveillance interval for slave relay testing is R.

* MODES 1, 2, 3, and 4.
During movement of recently irradiated fuel assemblies.

INSTRUMENTATION

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 required radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG 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

| FUNCTIONAL UNIT | CHANNELS TO TRIP/ALARM | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM/TRIP SETPOINT | ACTION |
|---|---|---|---|---|---|
| 1. Containment | | | | | |
| a. Deleted | | | | | |
| b. RCS Leakage Detection | | | | | |
| 1) Particulate Radioactivity | N.A. | 1 | 1, 2, 3, 4 | N.A. | 29 |
| 2) Deleted | | | | | |
| 2. Fuel Storage Pool Area Monitors | | | | | |
| a. Radiation Level | 1 | 2 | * | ≤ 15 mR/h | 28 |

## TABLE 3.3-6 (Continued)

## TABLE NOTATIONS

* With fuel in the fuel storage pool areas.

ACTION STATEMENTS

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, recently irradiated fuel assembly movement may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving movement of recently irradiated fuel assemblies.

ACTION 29 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

## TABLE 4.3-3
## RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| 1. Containment | | | | |
| a. Deleted | | | | |
| b. RCS Leakage Detection | | | | |
| 1) Particulate Radio-activity | SFCP | SFCP | SFCP | 1, 2, 3, 4 |
| 2) Deleted | | | | |
| 2. Fuel Storage Pool Area Monitors | | | | |
| a. Radiation Level | SFCP | SFCP | SFCP | * |

TABLE NOTATIONS

* With fuel in the fuel storage pool area.

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown Instrumentation transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 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 one or more Remote Shutdown Instrumentation transfer switches, power, or control circuits inoperable, 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 required 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 Instrumentation transfer switch, power and control circuit including the actuated components, shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program.

# TABLE 3.3-9

## REMOTE SHUTDOWN INSTRUMENTATION

| INSTRUMENT | READOUT LOCATION | TOTAL NO. OF CHANNELS | MINIMUM CHANNELS OPERABLE |
|---|---|---|---|
| 1. Reactor Trip Breaker Indication | Reactor Trip Switchgear | 1/trip breaker | 1/trip breaker |
| 2. Pressurizer Pressure | Aux. Shutdown Panel | 2 | 1 |
| 3. Pressurizer Level | Aux. Shutdown Panel | 2 | 1 |
| 4. Steam Generator Pressure | Aux. Shutdown Panel | 2/steam generator | 1/steam generator |
| 5. Steam Generator Water Level | Aux. Shutdown Panel | 2/steam generator | 1/steam generator |
| 6. Auxiliary Feedwater Flow Rate | Aux. Shutdown Panel | 1/steam generator | 1/steam generator |
| 7. Loop Hot Leg Temperature | Aux. Shutdown Panel | 1/loop | 1/loop |
| 8. Loop Cold Leg Temperature | Aux. Shutdown Panel | 1/loop | 1/loop |
| 9. Reactor Coolant System Pressure (Wide Range) | Aux. Shutdown Panel | 2 | 1 |
| 10. DWST Level | Aux. Shutdown Panel | 2 | 1 |
| 11. RWST Level | Aux. Shutdown Panel | 2 | 1 |
| 12. Containment Pressure | Aux. Shutdown Panel | 2 | 1 |
| 13. Emergency Bus Voltmeters | Aux. Shutdown Panel | 1/train | 1/train |
| 14. Source Range Count Rate | Aux. Shutdown Panel | 2 | 1 |
| 15. Intermediate Range Flux | Aux. Shutdown Panel | 2 | 1 |
| 16. Boric Acid Tank Level | Aux. Shutdown Panel | 2/tank | 1/tank |

| TRANSFER SWITCHES | SWITCH LOCATION |
|---|---|
| 1. Auxiliary Feedwater Isolation FWA*MOV35A | Transfer Switch Panel |
| 2. Auxiliary Feedwater Isolation FWA*MOV35B | Transfer Switch Panel |
| 3. Auxiliary Feedwater Isolation FWA*MOV35C | Transfer Switch Panel |
| 4. Auxiliary Feedwater Isolation FWA*MOV35D | Transfer Switch Panel |
| 5. Auxiliary Feedwater Pump Ah. Suction FWA*AOV23A | Transfer Switch Panel |
| 6. Auxiliary Feedwater Pump Ah. Suction FWA*AOV23B | Transfer Switch Panel |

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

| TRANSFER SWITCHES | SWITCH LOCATION |
|---|---|
| 7. Turbine Driven Pump Steam Supply MSS*AOV31A | Transfer Switch Panel |
| 8. Turbine Driven Pump Steam Supply MSS*AOV31B | Transfer Switch Panel |
| 9. Turbine Driven Pump Steam Supply MSS*AOV31D | Transfer Switch Panel |
| 10. Reactor Vessel Head Vent Isolation RCS*SV8095A | Transfer Switch Panel |
| 11. Reactor Vessel Head Vent Isolation RCS*SV8095B | Transfer Switch Panel |
| 12. Reactor Vessel Head Vent Isolation RCS*SV8096A | Transfer Switch Panel |
| 13. Reactor Vessel Head Vent Isolation RCS*SV8096B | Transfer Switch Panel |
| 14. Reactor Vessel to Excess Letdown RCS*MV8098 | Transfer Switch Panel |
| 15. Pressurizer Level Control RCS*LCV459 | Transfer Switch Panel |
| 16. Pressurizer Level Control RCS*LCV460 | Transfer Switch Panel |
| 17. Letdown Orifice Isolation CHS*AV8149A | Transfer Switch Panel |
| 18. Letdown Orifice Isolation CHS*AV8149B | Transfer Switch Panel |
| 19. Letdown Orifice Isolation CHS*AV8149C | Transfer Switch Panel |
| 20. Volume Control Tank Outlet Isolation CHS*LCV112B | Transfer Switch Panel |
| 21. Volume Control Tank Outlet Isolation CHS*LCV112C | Transfer Switch Panel |
| 22. RWST to CHS Pump Suction CHS*LCV112D | Transfer Switch Panel |
| 23. RWST to CHS Pump Suction CHS*LCV112E | Transfer Switch Panel |
| 24. Charging to RCS Isolation CHS*AV8146 | Transfer Switch Panel |
| 25. Charging to RCS Isolation CHS*AV8147 | Transfer Switch Panel |
| 26. Boric Acid Gravity Feed CHS*MV8507A | Transfer Switch Panel |
| 27. Boric Acid Gravity Feed CHS*MV8507B | Transfer Switch Panel |

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

| TRANSFER SWITCHES | SWITCH LOCATION |
|---|---|
| 28. Charging Header Isolation Bypass CHS*MV8116 | Transfer Switch Panel |
| 29. Pressurizer Heater Backup RCS*H1A (Group A) | Transfer Switch Panel |
| 30. Pressurizer Heater Backup RCS*H1B (Group B) | Transfer Switch Panel |

| CONTROL CIRCUITS | SWITCH LOCATION |
|---|---|
| 1. Auxiliary Feedwater Flow Control FWA*HV31A | Auxiliary Shutdown Panel |
| 2. Auxiliary Feedwater Flow Control FWA*HV31B | Auxiliary Shutdown Panel |
| 3. Auxiliary Feedwater Flow Control FWA*HV31C | Auxiliary Shutdown Panel |
| 4. Auxiliary Feedwater Flow Control FWA*HV31D | Auxiliary Shutdown Panel |
| 5. Auxiliary Feedwater Flow Control FWA*HV32A | Auxiliary Shutdown Panel |
| 6. Auxiliary Feedwater Flow Control FWA*HV32B | Auxiliary Shutdown Panel |
| 7. Auxiliary Feedwater Flow Control FWA*HV32C | Auxiliary Shutdown Panel |
| 8. Auxiliary Feedwater Flow Control FWA*HV32D | Auxiliary Shutdown Panel |
| 9. Auxiliary Feedwater Flow Control FWA*HV36A | Auxiliary Shutdown Panel |
| 10. Auxiliary Feedwater Flow Control FWA*HV36B | Auxiliary Shutdown Panel |
| 11. Auxiliary Feedwater Flow Control FWA*HV36C | Auxiliary Shutdown Panel |
| 12. Auxiliary Feedwater Flow Control FWA*HV36D | Auxiliary Shutdown Panel |

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

| CONTROL CIRCUITS | SWITCH LOCATION |
|---|---|
| 13. Reactor Vessel to PRT Control RCS*HCV442A | Auxiliary Shutdown Panel |
| 14. Reactor Vessel to PRT Control RCS*HCV442B | Auxiliary Shutdown Panel |
| 15. Charging Header Flow Control CHS*HCV190A | Auxiliary Shutdown Panel |
| 16. Charging Header Flow Control CHS*HCV190B | Auxiliary Shutdown Panel |
| 17. Excess Letdown Flow Control CHS*HCV123 | Auxiliary Shutdown Panel |
| 18. Charging Flow Control CHS*FCV121 | Auxiliary Shutdown Panel |
| 19. Low Pressure Letdown Control CHS*PCV131 | Auxiliary Shutdown Panel |

# TABLE 4.3-6
## REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|---|
| 1. | Reactor Trip Breaker Indication | SFCP | N.A. |
| 2. | Pressurizer Pressure | SFCP | SFCP |
| 3. | Pressurizer Level | SFCP | SFCP |
| 4. | Steam Generator Pressure | SFCP | SFCP |
| 5. | Steam Generator Water Level | SFCP | SFCP |
| 6. | Auxiliary Feedwater Flow Rate | SFCP | SFCP |
| 7. | Loop Hot Leg Temperature | SFCP | SFCP |
| 8. | Loop Cold Leg Temperature | SFCP | SFCP |
| 9. | Reactor Coolant System Pressure (Wide Range) | SFCP | SFCP |
| 10. | DWST Level | SFCP | SFCP |
| 11. | RWST Level | SFCP | SFCP |
| 12. | Containment Pressure | SFCP | SFCP |
| 13. | Emergency Bus Voltmeters | SFCP | SFCP |
| 14. | Source Range Count Rate | SFCP* | SFCP |
| 15. | Intermediate Range Amps | SFCP | SFCP |
| 16. | Boric Acid Tank Level | SFCP | SFCP |

* When below P-6 (intermediate range neutron flux interlock setpoint).

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:

a. With the number of OPERABLE accident monitoring instrumentation channels except the containment area high range radiation monitor, and reactor vessel water level, less than the Total Number of Channels shown in 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.

b. With the number of OPERABLE accident monitoring instrumentation channels except the containment area-high range radiation monitor, and reactor vessel water level less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours 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 channels for the containment area-high range radiation monitor less than required by either the total or the Minimum Channels OPERABLE requirements, initiate an 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 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. Deleted

e. With the number of OPERABLE channels for the reactor vessel water level monitor less than the Total number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the channel to OPERABLE status.

f. With the number of OPERABLE channels for the reactor vessel water level monitor less than the minimum channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the channel(s) to OPERABLE status; and

3. Restore the channel(s) to OPERABLE status at the next scheduled refueling.

SURVEILLANCE REQUIREMENTS

4.3.3.6.1 Each required 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.

4.3.3.6.2 Deleted

## TABLE 3.3-10

## ACCIDENT MONITORING INSTRUMENTATION

| INSTRUMENT | TOTAL NO. OF CHANNELS | MINIMUM CHANNELS OPERABLE |
|---|---|---|
| 1. Containment Pressure | | |
| a. Normal Range | 2 | 1 |
| b. Extended Range | 2 | 1 |
| 2. Reactor Coolant Outlet Temperature - $T_{HOT}$ (Wide Range) | 2 | 1 |
| 3. Reactor Coolant Inlet Temperature - $T_{COLD}$ (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 | 1/steam generator | 1/steam generator |
| 8. Steam Generator Water Level - Wide Range | 1/steam generator | 1/steam generator |
| 9. Refueling Water Storage Tank Water Level | 2 | 1 |
| 10. Demineralized Water Storage Tank Water Level | 2 | 1 |
| 11. Auxiliary Feedwater Flow Rate | 2/steam generator | 1/steam generator |
| 12. Reactor Coolant System Subcooling Margin Monitor | 2 | 1 |
| 13. Containment Water Level (Wide Range) | 2 | 1 |
| 14. Core Exit Thermocouples | 4/core quadrant | 2/core quadrant |
| 15. DELETED | | |

**TABLE 3.3-10 (Continued)**
**ACCIDENT MONITORING INSTRUMENTATION**

| INSTRUMENT | | TOTAL NO. OF CHANNELS | MINIMUM CHANNELS OPERABLE |
|---|---|---|---|
| 16. | Containment Area - High Range Radiation Monitor | 2 | 1 |
| 17. | Reactor Vessel Water Level | 2* | 1* |
| 18. | Deleted | | |
| 19. | Neutron Flux | 2 | 1 |

* A channel consists of eight sensors in a probe. A channel is OPERABLE if four or more sensors, half or more in the upper head region and half or more in the upper plenum region, are OPERABLE.

**TABLE 4.3-7**
**ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS**

| INSTRUMENT | | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|---|
| 1. | Containment Pressure | | |
| | a. Normal Range | SFCP | SFCP |
| | b. Extended Range | SFCP | SFCP |
| 2. | Reactor Coolant Outlet Temperature - $T_{HOT}$ (Wide Range) | SFCP | SFCP |
| 3. | Reactor Coolant Inlet Temperature - $T_{COLD}$ (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. | Demineralized Water Storage Tank Water Level | SFCP | SFCP |
| 11. | Auxiliary Feedwater Flow Rate | SFCP | SFCP |
| 12. | Reactor Coolant System Subcooling Margin Monitor | SFCP | SFCP |
| 13. | Containment Water Level (Wide Range) | SFCP | SFCP |
| 14. | Core Exit Thermocouples | SFCP | SFCP |
| 15. | DELETED | | |

**TABLE 4.3-7 (Continued)**
**ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS**

| INSTRUMENT | | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|---|
| 16. | Containment Area - High Range Radiation Monitor | SFCP | SFCP* |
| 17. | Reactor Vessel Water Level | SFCP | SFCP** |
| 18. | Deleted | | |
| 19. | Neutron Flux | SFCP | SFCP |

* 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.

** Electronic calibration from the ICC cabinets only.

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

THIS PAGE LEFT BLANK INTENTIONALLY

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

THIS PAGE LEFT BLANK INTENTIONALLY

## 3/4.3.5 SHUTDOWN MARGIN MONITOR

### LIMITING CONDITION FOR OPERATION

3.3.5 Two channels of Shutdown Margin Monitors shall be OPERABLE

a. With a minimum count rate as designated in the CORE OPERATING LIMITS REPORT (COLR), or

b. If the minimum count rate in Specification 3.3.5.a cannot be met, then the Shutdown Margin Monitors may be made OPERABLE with a lower minimum count rate, as specified in the COLR, by borating the Reactor Coolant System above the requirements of Specification 3.1.1.1.2 or 3.1.1.2. The additional boration shall be:

1. A minimum of 150 ppm above the SHUTDOWN MARGIN requirements specified in the COLR for MODE 3, or

2. A minimum of 350 ppm above the SHUTDOWN MARGIN requirements specified in the COLR for MODE 4, MODE 5 with RCS loops filled, and MODE 5 with RCS loops not filled.

APPLICABILITY: MODES 3*, 4, and 5.

ACTION:

a. With one Shutdown Margin Monitor inoperable, restore the inoperable channel to OPERABLE status within 48 hours.

b. With both Shutdown Margin Monitors inoperable or one Shutdown Margin Monitor inoperable for greater than 48 hours, immediately suspend all operations involving positive reactivity additions** via dilution and rod withdrawal. Verify the valves listed in Specification 4.1.1.2.2 are closed and secured in position within the next 4 hours and at least once per 14 days thereafter.*** Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1.2 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

---

* The shutdown margin monitors may be blocked during reactor startup in accordance with approved plant procedures.

** Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM.

*** The valves may be opened on an intermittent basis under administrative controls as noted in Surveillance 4.1.1.2.2.

SURVEILLANCE REQUIREMENTS

4.3.5 a. Each of the above required shutdown margin monitoring instruments shall be demonstrated OPERABLE by an ANALOG CHANNEL OPERATIONAL TEST at the frequency specified in the Surveillance Frequency Control Program that shall include verification that the Shutdown Margin Monitor is set per the CORE OPERATING LIMITS REPORT (COLR).

b. At the frequency specified in the Surveillance Frequency Control Program VERIFY the minimum count rate (counts/sec) as defined within the COLR.

## 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

### STARTUP AND POWER OPERATION

### LIMITING CONDITION FOR OPERATION

3.4.1.1 Four reactor coolant loops shall be OPERABLE and 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.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE, with at least three reactor coolant loops in operation when the Control Rod Drive System is capable of rod withdrawal or with at least one reactor coolant loop in operation when the Control Rod Drive System is not capable of rod withdrawal:*

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

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

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

d. Reactor Coolant Loop 4 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 less than the above required reactor coolant loops in operation and the Control Rod Drive System is capable of rod withdrawal, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1.2 and immediately initiate corrective action to return the required reactor coolant loop to operation.

* All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1.1.2, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% 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 Either: *, **

a. With the Control Rod Drive System capable of rod withdrawal, at least two RCS loops shall be OPERABLE and in operation, or

b. With the Control Rod Drive System not capable of rod withdrawal, at least two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and at least one of these loops shall be in operation. For RCS loop(s) to be OPERABLE, at least one reactor coolant pump (RCP) shall be in operation.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

---

* All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1.1.2, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** The first reactor coolant pump shall not be started when any RCS loop wide range cold leg temperature is ≤ 226°F unless:

a. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

b. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

## 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 17%.

APPLICABILITY: MODE 5 with at least two reactor coolant loops filled***.

---

*a. The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1.1.2, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

b. All RHR loops may be removed from operation during a planned heatup to MODE 4 when at least one RCS loop is OPERABLE and in operation and when two additional steam generators are OPERABLE as required by LCO 3.4.1.4.1.b.

** 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 first reactor coolant pump shall not be started when:

a. Any RCS loop wide range cold leg temperature is > 150°F unless:

1. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

2. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

b. All RCS loop wide range cold leg temperatures are ≤ 150°F unless the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

ACTION:

a. With less than the required RHR loop(s) OPERABLE or 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 operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1.2 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 The required pump, if not in operation, shall be determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

4.4.1.4.1.4 Locations susceptible to gas accumulation in the required RHR trains shall be verified to be sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

COLD SHUTDOWN - LOOPS NOT FILLED

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 less than two reactor coolant loops 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 operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1.2 and immediately initiate corrective action to return the required RHR loop to operation.

---

* 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 introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1.1.2, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

*** The first reactor coolant pump shall not be started when:

a. Any RCS loop wide range cold leg temperature is > 150°F unless:

   1. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

   2. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

b. All RCS loop wide range cold leg temperatures are ≤ 150°F unless the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 The required pump, if not in operation, shall be determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

4.4.1.4.2.2 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.

4.4.1.4.2.3 Locations susceptible to gas accumulation in the required RHR trains shall be verified to be sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

LOOP STOP VALVES

LIMITING CONDITION FOR OPERATION

3.4.1.5 Each RCS loop stop valve shall be open and the power removed from the valve operator.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With power available to one or more loop stop valve operators, remove power from the loop stop valve operators within 30 minutes or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b.*(1) With one or more RCS loop stop valves closed, maintain the valve(s) closed and be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.5 Verify each RCS loop stop valve is open and the power removed from the valve operator at the frequency specified in the Surveillance Frequency Control Program.

*(1)All required ACTIONS of ACTION Statement 3.4.1.5.b shall be completed whenever this action is entered.

REACTOR COOLANT SYSTEM

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated with power removed from the associated RCS loop stop valve operators until:

a. The temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops, and

b. The boron concentration of the isolated loop is greater than or equal to the boron concentration required by Specifications 3.1.1.1.2 or 3.1.1.2 for MODE 5 or Specification 3.9.1.1 for MODE 6.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the requirements of the above specification not satisfied, do not open the isolated loop stop valves.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.2 The isolated loop boron concentration shall be determined to be greater than or equal to the boron concentration required by Specifications 3.1.1.1.2 or 3.1.1.2 for MODE 5 or Specification 3.9.1.1 for MODE 6 within 2 hours prior to opening the hot or cold leg stop valve.

## 3/4.4.2 SAFETY VALVES

### LIMITING CONDITION FOR OPERATION

3.4.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting* of 2500 psia ± 3%.**

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 226°F.

ACTION:

With one pressurizer Code safety valve inoperable, restore the inoperable valve to OPERABLE status within 15 minutes. If the inoperable valve is not restored to OPERABLE status within 15 minutes, or if two or more pressurizer Code safety valves are inoperable, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with any RCS cold leg temperature ≤ 226°F within the following 24 hours.

### SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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

**The lift setting shall be within ± 1% following pressurizer Code safety valve testing.

This Page Intentionally Left Blank

## 3/4.4.3 PRESSURIZER

## STARTUP AND POWER OPERATION

## LIMITING CONDITION FOR OPERATION

3.4.3.1 The pressurizer shall be OPERABLE with:

a. at least two groups of pressurizer heaters, each having a capacity of at least 175 kW; and

b. water level maintained at programmed level +/-6% of full scale (Figure 3.4-5).

APPLICABILITY: MODES 1 and 2.

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.

b. With pressurizer water level outside the parameters described in Figure 3.4-5, within 2 hours restore programmed level to within +/- 6% of full scale, or be in at least HOT STANDBY within the next 6 hours.

c. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours.

## SURVEILLANCE REQUIREMENTS

4.4.3.1.1 The pressurizer water level shall be verified to be within programmed level +/- 6% of full scale at the frequency specified in the Surveillance Frequency Control Program.

4.4.3.1.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at the frequency specified in the Surveillance Frequency Control Program.

PRESSURIZER LEVEL CONTROL

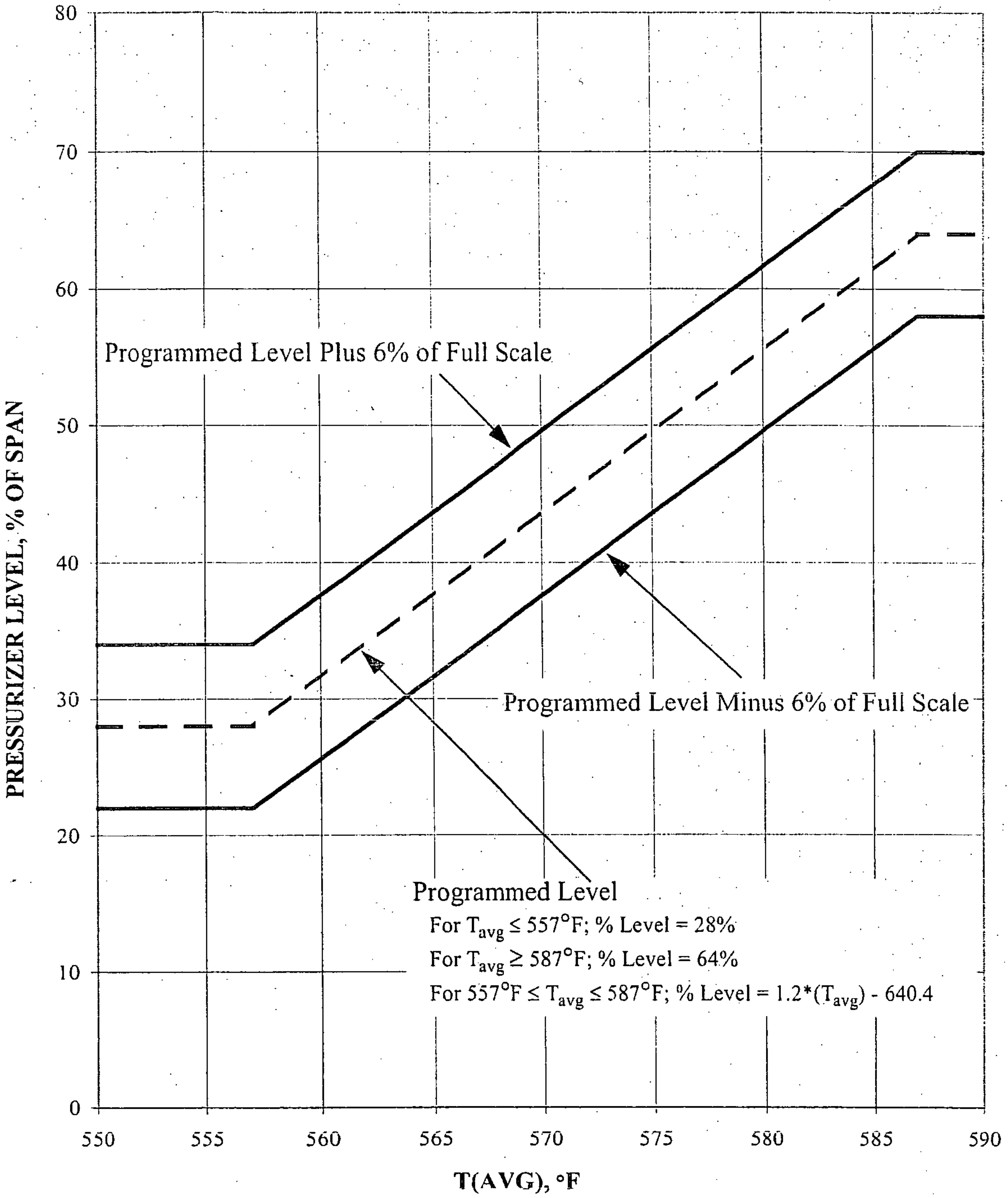



**FIGURE 3.4-5**

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.3.2 The pressurizer shall be OPERABLE with:

a. at least two groups of pressurizer heaters, each having a capacity of at least 175 kW; and

b. water level less than or equal to 89% of full scale.

APPLICABILITY: MODE 3

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours of being declared inoperable, or be in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in HOT SHUTDOWN within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The pressurizer water level shall be determined to be less than or equal to 89% of full scale at the frequency specified in the Surveillance Frequency Control Program.

4.4.3.2.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at the frequency specified in the Surveillance Frequency Control Program.

## 3/4.4.4 RELIEF VALVES

### LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to OPERABLE status, or place its associated PORV(s) control switch to “CLOSE.” Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program; and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4 at the frequency specified in the Surveillance Frequency Control Program; and

c. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV high pressurizer pressure actuation channels, but excluding valve operation, at the frequency specified in the Surveillance Frequency Control Program; and

d. Verify the PORV high pressure automatic opening function is enabled at the frequency specified in the Surveillance Frequency Control Program.

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 in order to meet the ACTION requirements of Specification 3.4.4.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam Generator (SG) tube integrity shall be maintained.

AND

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

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

ACTION:

NOTE

Separate ACTION entry is allowed for each SG tube.

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

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

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

b. With required ACTION and associated completion time of ACTION a. not met or SG tube integrity not maintained:

   1. Be in HOT STANDBY within 6 hours, and

   2. Be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

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

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

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

## 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

### LEAKAGE DETECTION SYSTEMS

### LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System, and

b. The Containment Drain Sump Monitoring System

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

ACTION:

a. With the Containment Atmosphere Particulate Radioactivity Monitor inoperable, operations may continue for up to 30 days provided:

   1. Appropriate grab samples of the containment atmosphere are obtained and analyzed for particulate radioactivity at least once per 24 hours, or

   2. A Reactor Coolant System water inventory balance is performed at least once per 24 hours during steady state operation.

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

b. With the Containment Drain Sump Monitoring System inoperable, operation may continue for up to 30 days provided:

   1. A Reactor Coolant System water inventory balance is performed at least once per 24 hours during steady state operation.

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

c. With the Containment Atmosphere Particulate Radioactivity Monitor inoperable and the Containment Drain Sump Monitoring System inoperable, operation may continue for up to 72 hours provided:

   1. Immediate action is initiated to restore either the Containment Atmosphere Particulate Radioactivity Monitor or the Containment Drain Sump Monitoring System to OPERABLE status, and

## 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

## LEAKAGE DETECTION SYSTEMS

## LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. Appropriate grab samples of the containment atmosphere are obtained and analyzed for particulate radioactivity within 6 hours and at least once per 6 hours thereafter, and

3. A Reactor Coolant System water inventory balance is performed within 6 hours and at least once per 6 hours thereafter.

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

## SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Particulate Radioactivity Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment Drain Sump Monitoring System-performance of CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program.

OPERATIONAL LEAKAGE

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,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2250 ± 20 psia, and

f.* 0.5 gpm LEAKAGE per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

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

b. With primary to secondary LEAKAGE not within limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any RCS operational LEAKAGE not within limits for reasons other than PRESSURE BOUNDARY LEAKAGE, primary to secondary LEAKAGE, or LEAKAGE from Reactor Coolant System Pressure Isolation Valves, reduce LEAKAGE to within limits within 4 hours or be in at least HOT STANDY within the next 6 hours and in the COLD SHUTDOWN within the following 30 hours.

---

* This requirement does not apply to Pressure Isolation Valves in the Residual Heat Removal flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

d. With any Reactor Coolant System Pressure Isolation Valve LEAKAGE greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least 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.

4.4.6.2.1 Reactor Coolant System operational LEAKAGE shall be demonstrated to be within each of the above limits by:

a. Deleted

b. Deleted

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2250 ±20 psia at the frequency specified in the Surveillance Frequency Control Program with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

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

1. Not required to be performed until 12 hours after establishment of steady state operation.
2. Not applicable to primary to secondary LEAKAGE.

---

d. Performance of a Reactor Coolant System water inventory balance at the frequency specified in the Surveillance Frequency Control Program;

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

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

---

e. Verification that primary to secondary LEAKAGE is ≤ 150 gallons per day through any one Steam Generator at the frequency specified in the Surveillance Frequency Control Program, and;

f. Monitoring the Reactor Head Flange Leakoff System at the frequency specified in the Surveillance Frequency Control Program.

4.4.6.2.2[(1)(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. At the frequency specified in the Surveillance Frequency Control Program,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Deleted

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, and

e. When tested pursuant to Specification 4.0.5.

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

(2) This surveillance is not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the RHR flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

## TABLE 3.4-1

## REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

| VALVE NUMBER | FUNCTION |
|---|---|
| 3-SIL-V15 | SI Tank 1A Discharge Isolation Valve |
| 3-SIL-V17 | SI Tank 1B Discharge Isolation Valve |
| 3-SIL-V19 | SI Tank 1C Discharge Isolation Valve |
| 3-SIL-V21 | SI Tank 1D Discharge Isolation Valve |
| 3-SIL-V26 | RHR/SI to RCS Loop 2, Hot Leg |
| 3-SIL-V27 | SIH to RCS Loop 2, Hot Leg |
| 3-SIL-V28 | RHR/SI to RCS Loop 4, Hot Leg |
| 3-SIL-V29 | SIH to RCS Loop 4, Hot Leg |
| 3-SIL-V984 | RHR/SI to RCS Loop 4, Cold Leg |
| 3-SIL-V985 | RHR/SI to RCS Loop 3, Cold Leg |
| 3-SIL-V986 | RHR/SI to RCS Loop 2, Cold Leg |
| 3-SIL-V987 | RHR/SI to RCS Loop 1, Cold Leg |
| 3-SIH-V5 | SIH to RCS Cold Legs |
| 3-SIH-V110 | SIH to RCS Loop 1, Hot Leg |
| 3-SIH-V112 | SIH to RCS Loop 3, Hot Leg |
| 3-RCS-V26 | SIH to RCS Loop 1, Hot Leg |
| 3-RCS-V29 | SIH to RCS Loop 1, Cold Leg |
| 3-RCS-V30 | SIL to RCS Loop 1, Cold Leg |
| 3-RCS-V69 | RHR/SI to RCS Loop 2, Hot Leg |
| 3-RCS-V70 | SIH to RCS Loop 2, Cold Leg |
| 3-RCS-V71 | SIL to RCS Loop 2, Cold Leg |
| 3-RCS-V102 | SIH to RCS Loop 3, Hot Leg |
| 3-RCS-V106 | SIH to RCS Loop 3, Cold Leg |
| 3-RCS-V107 | SIL to RCS Loop 3, Cold Leg |
| 3-RCS-V142 | RHR/SI to RCS Loop 4, Hot Leg |
| 3-RCS-V145 | SIH to RCS Loop 4, Cold Leg |
| 3-RCS-V146 | SIL to RCS Loop 4, Cold Leg |
| 3-RHS-MV8701C | RCS Loop 1, Hot Leg to RHR |
| 3-RHS-MV8702C | RCS Loop 4, Hot Leg to RHR |
| 3-RHS-MV8701A | RCS Loop 1, Hot Leg to RHR |
| 3-RHS-MV8702B | RCS Loop 4, Hot Leg to RHR |

This page intenionally left blank

This page intentionally left blank

This page intentionally left blank

## 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 81.2 microCuries per gram DOSE EQUIVALENT XE-133.

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

ACTION:

a. With the specific activity of the reactor coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 less than or equal to 60 microCuries per gram once per 4 hours.

b. With the specific activity of the reactor coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 but less than or equal to 60 microCuries per gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 microCurie per gram limit. Specification 3.0.4.c is applicable.

c. With the specific activity of the reactor coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or greater than 60 microCuries per gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within 36 hours.

d. With the specific activity of the reactor coolant greater than 81.2 microCuries per gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 81.2 microCuries per gram limit. Specification 3.0.4.c is applicable.

e. With the specific activity of the reactor coolant greater than 81.2 microCuries per gram DOSE EQUIVALENT XE-133 for more than 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1 Verify the specific activity of the reactor coolant less than or equal to 81.2 microCuries per gram DOSE EQUIVALENT XE-133 at the frequency specified in the Surveillance Frequency Control Program.*

4.4.8.2 Verify the specific activity of the reactor coolant less than or equal to 1.0 microCuries per gram DOSE EQUIVALENT I-131 at the frequency specified in the Surveillance Frequency Control Program,* and between 2 and 6 hours after a THERMAL POWER change of greater than or equal to 15% RATED THERMAL POWER within a one hour period.

* Surveillance only required to be performed for MODE 1 operation, consistent with the provisions of Specification 4.0.1.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

## 3/4.4.9 PRESSURE/TEMPERATURE LIMITS

### LIMITING CONDITION FOR OPERATION

3.4.9.1 Reactor Coolant System (except the pressurizer) temperature, pressure, and heatup and cooldown rates of ferritic materials shall be limited in accordance with the limits shown on Figures 3.4-2 and 3.4-3. In addition, a maximum of one reactor coolant pump can be in operation when the lowest unisolated Reactor Coolant System loop wide range cold leg temperature is ≤ 160°F.

APPLICABILITY: At all times.

ACTION:

a. With any of the above limits exceeded in MODES 1, 2, 3, or 4, perform the following:

   1. Restore the temperature and/or pressure to within limit within 30 minutes.

   AND

   2. Perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System remains acceptable for continued operation within 72 hours. Otherwise, be in at least MODE 3 within the next 6 hours and in MODE 5 with RCS pressure less than 500 psia within the following 30 hours.

b. With any of the above limits exceeded in other than MODES 1, 2, 3, or 4, perform the following:

   1. Immediately initiate action to restore the temperature and/or pressure to within limit.

   AND

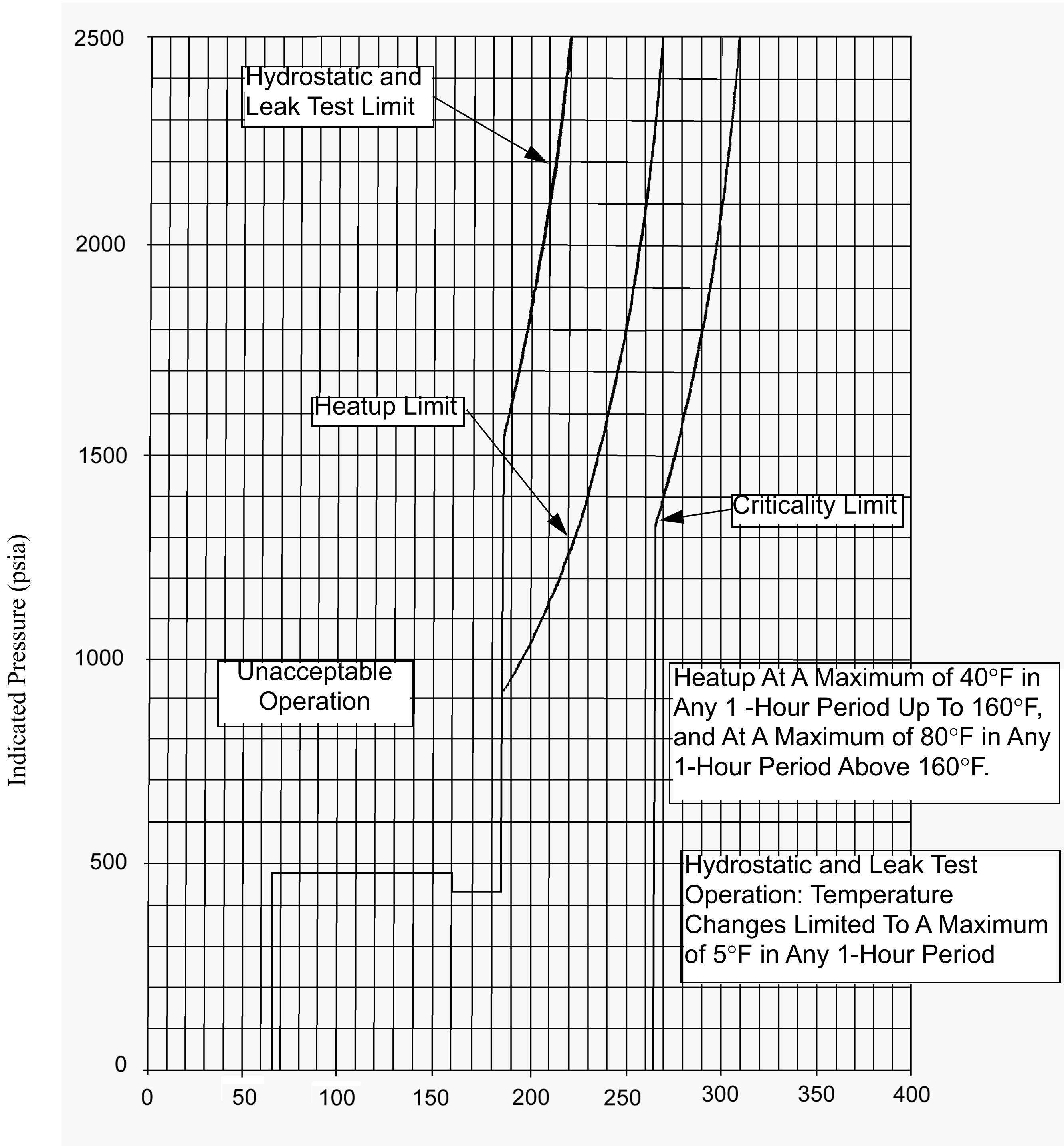
   2. Perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System is acceptable for continued operation prior to entering MODE 4.

### SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup and cooldown operations, and during the one-hour period prior to and during inservice leak and hydrostatic testing operations.

4.4.9.1.2 DELETED

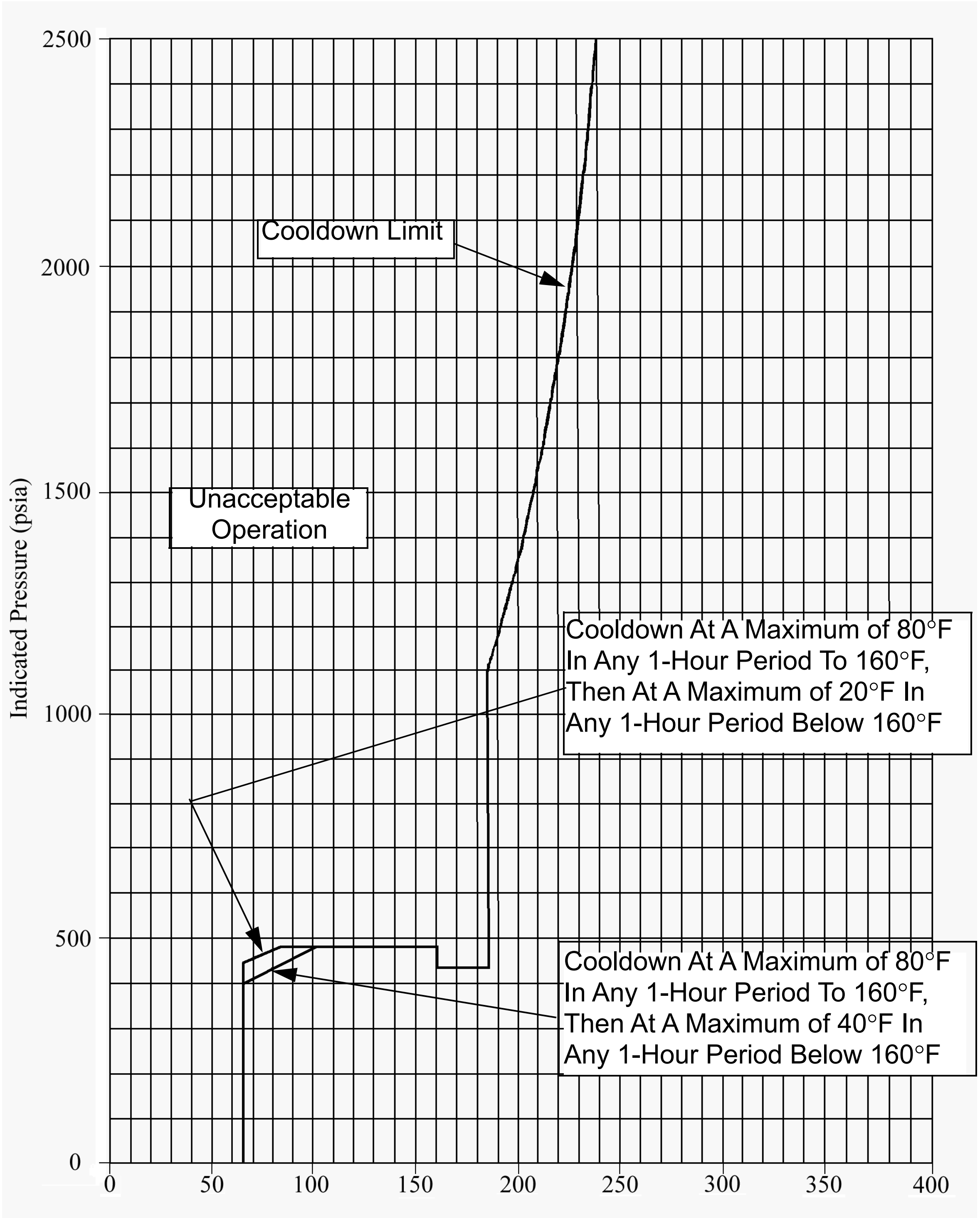
**Heatup Limitations for Fluence up to 2.72E+19 n/cm$^2$ (54 EFPY)**




**Indicated Cold Leg Temperature (°F)**

**Figure 3.4-2**

**Cooldown Limitations for Fluence up to 2.72E+19 n/cm$^2$ (54 EFPY)**




**Indicated Cold Leg Temperature (°F)**

**Figure 3.4-3**

THIS PAGE INTENTIONALLY LEFT BLANK

This page intentionally left blank

LIMITING CONDITION FOR OPERATION

3.4.9.3 Cold Overpressure Protection shall be OPERABLE with a maximum of one centrifugal charging pump* and no Safety Injection pumps capable of injecting into the Reactor Coolant System (RCS) and one of the following pressure relief capabilities:

1. One power operated relief valve (PORV) with a nominal lift setting established in Figure 3.4-4a and one PORV with a nominal lift setting established in Figure 3.4-4b with no more than one isolated RCS loop, or

2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 426.8 psig and ≤453.2 psig, or

3. One PORV with a nominal lift setting established in Figure 3.4-4a or Figure 3.4-4b with no more than one isolated RCS loop and one RHR suction relief valve with a setpoint ≥426.8 psig and ≤ 453.2 psig, or

4. RCS depressurized with an RCS vent of ≥ 2.0 square inches.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ 226°F, MODE 5, and MODE 6 when the head is on the reactor vessel.

ACTION:

– – – – – – – – – – – – – – – – – – NOTE – – – – – – – – – – – – – – – – – –

LCO 3.0.4.b is not applicable when entering MODE 4

– – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – –

a. With two or more centrifugal charging pumps capable of injecting into the RCS, immediately initiate action to establish that a maximum of one centrifugal charging pump is capable of injecting into the RCS.

b. With any Safety Injection pump capable of injecting into the RCS, immediately initiate action to establish that no Safety Injection pumps are capable of injecting into the RCS.

c. With one required relief valve inoperable in MODE 4, restore the required relief valve to OPERABLE status within 7 days, or depressurize and vent the RCS through at least a 2.0 square inch vent within the next 12 hours.

* Two centrifugal charging pumps may be capable of injecting into the RCS for less than one hour, during pump swap operations. However, at no time will two charging pumps be simultaneously out of pull-to-lock during pump swap operations.

d. With one required relief valve inoperable in MODE 5 or 6, restore the required relief valve to OPERABLE status within 24 hours, or depressurize the RCS and establish an RCS vent of $\geq$2.0 square inches within the next 12 hours.

e. With two required relief valves inoperable, depressurize the RCS and establish an RCS vent of $\geq$2.0 square inches within 12 hours.

f. In the event the PORVs, the RHR suction relief valves, or the RCS vent are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, the RHR suction relief valves, or RCS vent on the transient, and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Demonstrate that each required PORV is OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at the frequency specified in the Surveillance Frequency Control Program thereafter 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 block valve is open and the PORV Cold Overpressure Protection System (COPPS) is armed at the frequency specified in the Surveillance Frequency Control Program when the PORV is being used for overpressure protection.

4.4.9.3.2 Demonstrate that each required RHR suction relief valve is OPERABLE by:

a. Verifying the isolation valves between the RCS and each required RHR suction relief valve are open at the frequency specified in the Surveillance Frequency Control Program; and

b. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 When complying with 3.4.9.3.4, verify that the RCS is vented through a vent pathway $\geq$ 2.0 square inches at the frequency specified in the Surveillance Frequency Control Program for a passive vent path and at the frequency specified in the Surveillance Frequency Control Program for unlocked open vent valves.

4.4.9.3.4 Verify that no Safety Injection pumps are capable of injecting into the RCS at the frequency specified in the Surveillance Frequency Control Program.

4.4.9.3.5 Verify that a maximum of one centrifugal charging pump is capable of injecting into the RCS at the frequency specified in the Surveillance Frequency Control Program.

## High Setpoint PORV Curve
## For the Cold Overpressure Protection System

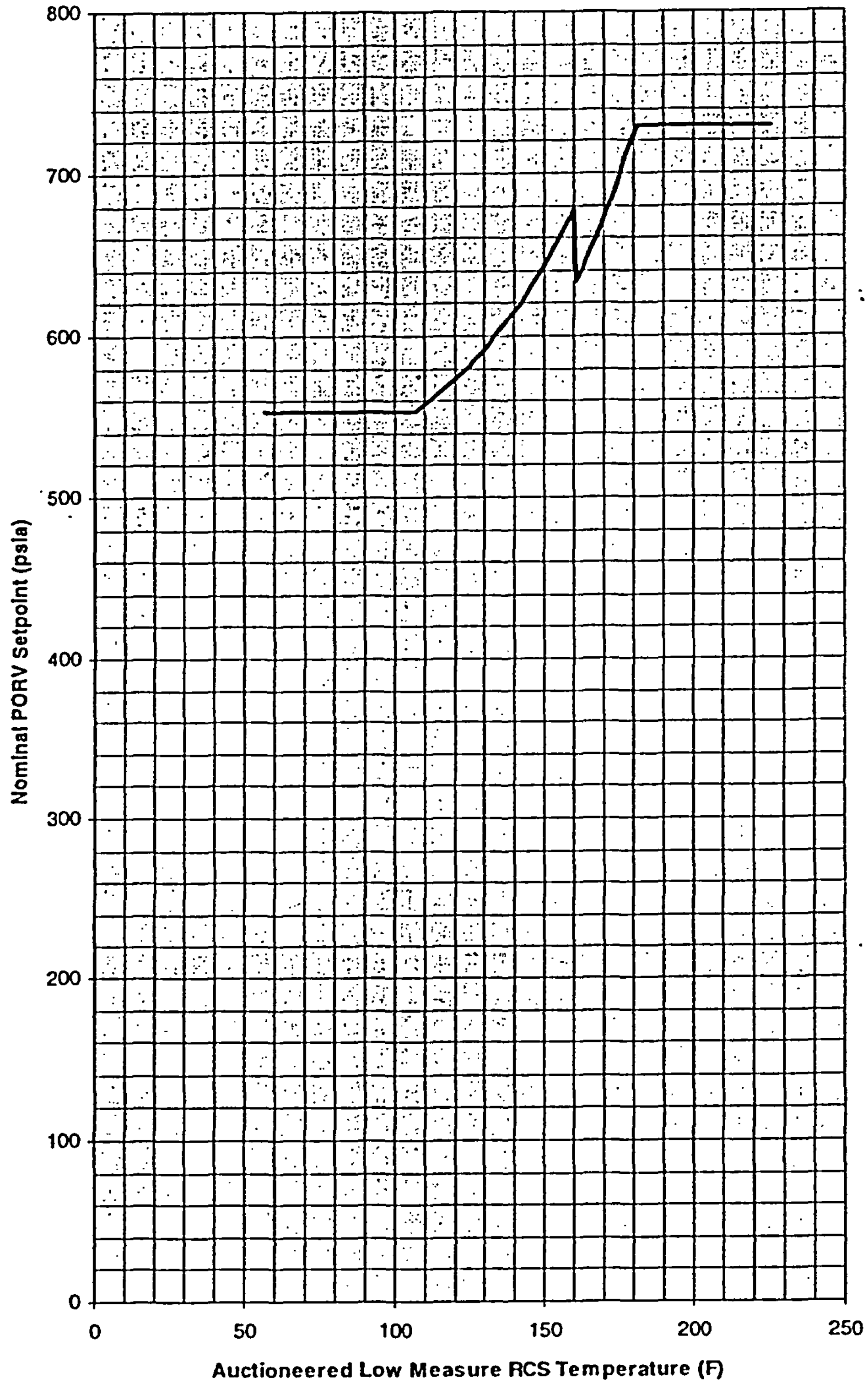



FIGURE 3.4-4a

Low Setpoint PORV Curve
For the Cold Overpressure Protection System

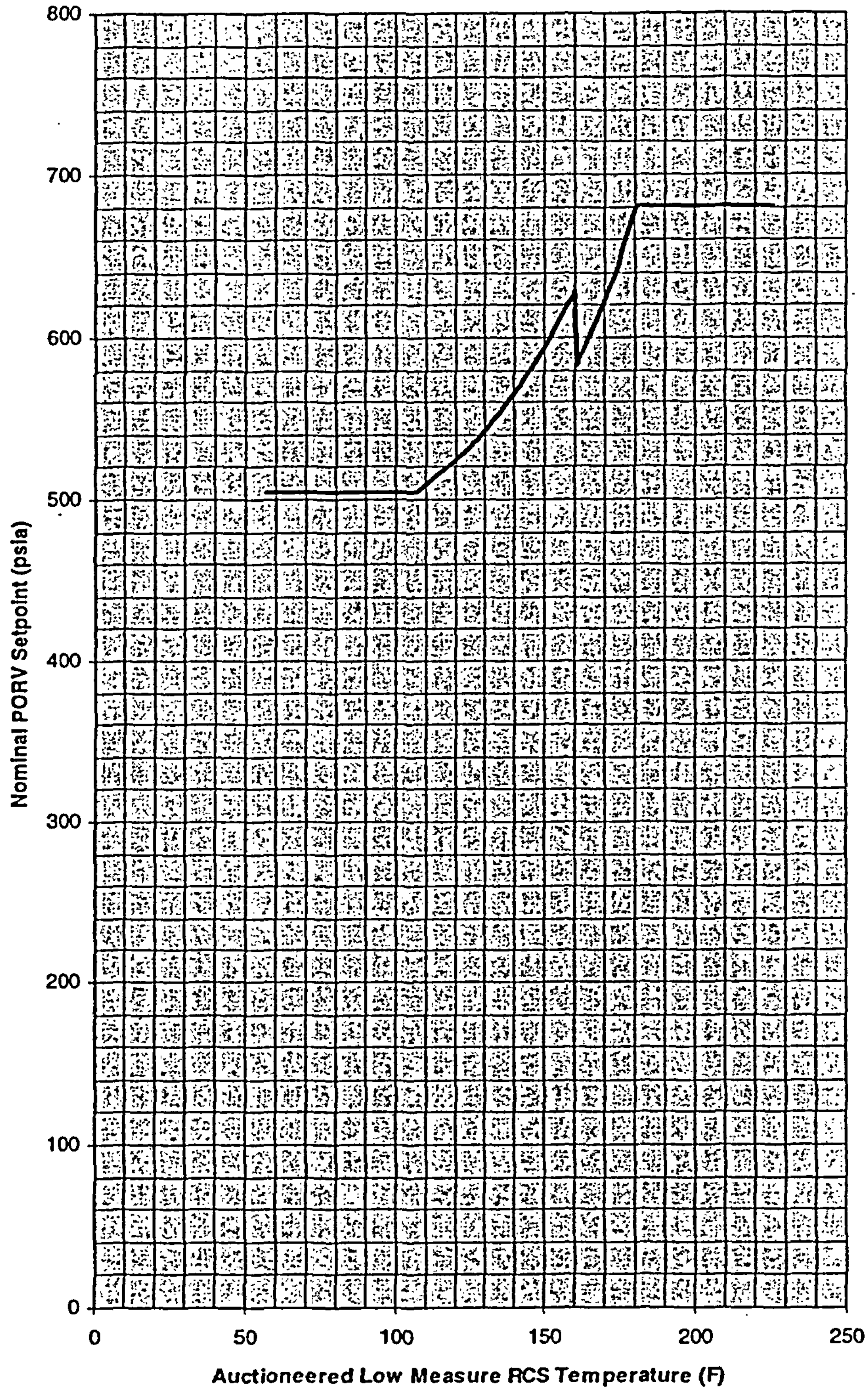



FIGURE 3.4-4b

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS

### 3/4.5.1 ACCUMULATORS

### LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6618 and 7030 gallons,

c. A boron concentration of between 2600 and 2900 ppm, and

d. A nitrogen cover-pressure of between 636 and 694 psia.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 8 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.

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.

### SURVEILLANCE REQUIREMENTS

4.5.1 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.

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 1% of tank volume by verifying the boron concentration of the accumulator solution. This surveillance is not required when the volume increase makeup source is the RWST.

---

* Pressurizer pressure above 1000 psig.

c. At the frequency specified in the Surveillance Frequency Control Program when the RCS pressure is above 1000 psig by verifying that the associated circuit breakers are locked in a deenergized position or removed.

## 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 centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,*

d. One OPERABLE RHR pump,*

e. One OPERABLE containment recirculation heat exchanger,

f. One OPERABLE containment recirculation pump, and

g. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and capable of automatically stopping the RHR pump and being manually realigned to transfer 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 be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*The allowable outage time for each RHR pump/RHR heat exchanger may be extended to 120 hours for the purpose of pump modification to change mechanical seal and other related modifications. This exception may only be used one time per RHR pump/RHR heat exchanger and is not valid after April 30, 1995.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the valve operators removed:

| Valve Number | Valve Function | Valve Position |
|---|---|---|
| 3SIH*MV8806 | RWST Supply to SI Pumps | OPEN |
| 3SIH*MV8802A | SI Pump A to Hot Leg Injection | CLOSED |
| 3SIH*MV8802B | SI Pump B to Hot Leg Injection | CLOSED |
| 3SIH*MV8835 | SI Cold Leg Master Isolation | OPEN |
| 3SIH*MV8813 | SI Pump Master Miniflow Isolation | OPEN |
| 3SIL*MV8840 | RHR to Hot Leg Injection | CLOSED |
| 3SIL*MV8809A | RHR Pump A to Cold Leg Injection | OPEN |
| 3SIL*MV8809B | RHR Pump B to Cold Leg Injection | OPEN |

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

1) Verifying that the ECCS piping locations susceptible to gas accumulation, except for the operating centrifugal charging pump(s) and associated piping, the RSS pump, the RSS heat exchanger and associated piping, are sufficiently filled with water, and

NOTE

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

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) At least once daily of the areas affected (during each day) within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

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 signal greater than or equal to 412.5 psia 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 (strainers, 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 a Safety Injection actuation test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

3) Verifying that the Residual Heat Removal pumps stop automatically upon receipt of a Low-Low RWST Level test signal.

f. By verifying that each of the following pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump

2) Safety Injection pump

3) RHR pump

4) Containment recirculation pump

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation when the ECCS subsystems are required to be OPERABLE, and

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

ECCS Throttle Valves

| Valve Number | Valve Number |
|---|---|
| 3SIH*V6 | 3SIH*V25 |
| 3SIH*V7 | 3SIH*V27 |

ECCS Throttle Valves

| Valve Number | Valve Number |
|---|---|
| 3SIH*V8 | 3SIH*V107 |
| 3SIH*V9 | 3SIH*V108 |
| 3SIH*V21 | 3SIH*V109 |
| 3SIH*V23 | 3SIH*V111 |

h. Deleted

## 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 centrifugal charging pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump,

d. One OPERABLE containment recirculation heat exchanger,

e. One OPERABLE containment recirculation pump, and

f. An OPERABLE flow path which, with manual realignment of valves, is capable of discharging to the RCS, taking suction from the refueling water storage tank, 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 pressure safety injection subsystem

---

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

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

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

## SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2, with the exception that valves may be out of alignment but capable of being manually realigned.

EMERGENCY CORE COOLING SYSTEMS

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 contained borated water volume between 1,166,000 and 1,207,000 gallons,

b. A boron concentration between 2700 and 2900 ppm of boron,

c. A minimum solution temperature of 42°F, and

d. A maximum solution temperature of 73°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 the next 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.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.5 pH TRISODIUM PHOSPHATE STORAGE BASKETS

LIMITING CONDITION FOR OPERATION

3.5.5 The trisodium phosphate (TSP) dodecahydrate Storage Baskets shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With the TSP Storage Baskets inoperable, restore the system TSP Storage Baskets to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The TSP Storage Baskets shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying that a minimum total of 974 cubic feet of TSP is contained in the TSP Storage Baskets.

## 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[(1)] not capable of being closed by OPERABLE containment automatic isolation valves,[(2)] and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions,[(3)] except for valves that are open under administrative control as permitted by Specification 3.6.3; and

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

c. Deleted

---

(1) 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.

(2) In MODE 4, the requirement for an OPERABLE containment isolation valve system is satisfied by use of the containment isolation actuation pushbuttons.

(3) Isolation devices in high radiation areas may be verified by use of administrative means.

## CONTAINMENT LEAKAGE

### LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

With the containment leakage rates exceeding the limits, restore the leakage rates to within limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

This page intentionally left blank

This page intentionally left blank.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 The containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

| NOTE |
| --- |
| Entry and exit through the containment air lock doors is permitted to perform repairs on the affected air lock components. |

a. With only one containment air lock door inoperable:

1. Verify the OPERABLE air lock door is closed within 1 hour and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

2. Operation may then continue provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,

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

b. With only the containment air lock interlock mechanism inoperable, verify an OPERABLE air lock door is closed within 1 hour and lock an OPERABLE air lock door closed within 24 hours. Verify an OPERABLE air lock door is locked closed at least once per 31 days thereafter. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (Entry into and exit from containment is permissible under the control of a dedicated individual).

LIMITING CONDITION FOR OPERATION

Continued

c. With the containment air lock inoperable, except as specified in ACTION a. or ACTION b. above, immediately initiate action to evaluate overall containment leakage rate per Specification 3.6.1.2 and verify an air lock door is closed within 1 hour. Restore the air lock 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.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage results in accordance with the Containment Leakage Rate Testing Program. Containment air lock leakage test results shall be evaluated against the leakage limits of Technical Specification 3.6.1.2. (An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test).

b. Deleted

c. At the frequency specified in the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

CONTAINMENT SYSTEMS

CONTAINMENT PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment pressure shall be maintained greater than or equal to 10.6 psia and less than or equal to 14.0 psia.

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

ACTION:

With the containment pressure less than 10.6 psia or greater than 14.0 psia, restore the containment 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 pressure shall be determined to be within the limits at the frequency specified in the Surveillance Frequency Control Program.

This Page Intentionally Left Blank

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained greater than or equal to 80°F and less than or equal to 120°F.

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

ACTION:

With the containment average air temperature less than 80°F or greater than 120°F, restore 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:

Location

a. 94 ft elevation, E outside crane wall

b. 86 ft elevation, NW outside crane wall

c. 75 ft elevation, W Steam Generator platform

d. 75 ft elevation, E Steam Generator platform

e. 45 ft elevation, Pressurizer cubicle, crane wall

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits 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 Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined at the frequency specified in the Containment Leakage Rate Testing Program.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and each 42-inch containment shutdown purge supply and exhaust isolation valve shall be closed and locked closed.

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

ACTION:

a. With a 42-inch containment purge supply and/or exhaust isolation valve open or not locked closed, close and/or lock 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.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The containment purge supply and exhaust isolation valves shall be verified to be locked closed and closed at the frequency specified in the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Quench Spray subsystems shall be OPERABLE.

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

ACTION:

With one Containment Quench Spray subsystem inoperable, restore the inoperable 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

4.6.2.1 Each Containment Quench Spray subsystem shall be demonstrated OPERABLE:

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

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

    2) Verifying the temperature of the borated water in the refueling water storage tank is between 42°F and 73°F.

b. By verifying that each pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5;

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 CDA test signal, and

    2) Verifying that each spray pump starts automatically on a CDA test signal.

d. By verifying each spray nozzle is unobstructed following maintenance that could cause nozzle blockage.

RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent Recirculation Spray Systems shall be OPERABLE.

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

ACTION:

With one Recirculation Spray System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Recirculation Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each Recirculation 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 each pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5;

c. At the frequency specified in the Surveillance Frequency Control Program by verifying that on a CDA test signal, each recirculation spray pump starts automatically after receipt of an RWST Low-Low signal;

d. 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 CDA test signal; and

e. By verifying each spray nozzle is unobstructed following maintenance that could cause nozzle blockage.

Intentionally Left Blank

## 3/4.6.3 CONTAINMENT ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE. [(1)] [(2)]

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation barrier OPERABLE in the affected penetration(s), and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate the affected penetration(s) within 4 hours by use of deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of closed manual valve(s) or blind flange(s); or

d. Isolate the affected penetration that has only one containment isolation valve and a closed system within 72 hours by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange; or

e. 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 DELETED

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE 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 High Radiation test signal, each purge supply and exhaust 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 when tested pursuant to Specification 4.0.5.

---

(1) The provisions of this Specification are not applicable for main steam line isolation valves. However, provisions of Specification 3.7.1.5 are applicable for main steam line isolation valves.

(2) Containment isolation valves may be opened on an intermittent basis under administrative controls.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

CONTAINMENT SYSTEMS

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

LIMITING CONDITION FOR OPERATION

3.6.5.1 The inside and outside isolation valves in the steam jet air ejector suction line shall be closed.

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

ACTION:

With the inside or outside isolation valves in the steam jet air ejector suction line not closed, restore the valve to the closed position 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

4.6.5.1.1 The steam jet air ejector suction line outside isolation valve shall be determined to be in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 200°F and at the frequency specified in the Surveillance Frequency Control Program.

4.6.5.1.2 The steam jet air ejector suction line inside isolation valve shall be determined to be locked in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 200°F.

## 3/4.6.6 SECONDARY CONTAINMENT

## SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Supplementary Leak Collection and Release Systems shall be OPERABLE with each system comprised of:

a. one OPERABLE filter and fan, and

b. one OPERABLE Auxiliary Building Filter System as defined in Specification 3.7.9.

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

ACTION:

With one Supplementary Leak Collection and Release 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.

### SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Supplementary Leak Collection and Release 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 a system flow rate of 7600 cfm to 9800 cfm and that the system operates for at least 15 continuous minutes with the heaters operating.

b. At the frequency specified in the Surveillance Frequency Control Program and following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the 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 system flow rate is 7600 cfm to 9800 cfm;

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F) and a relative humidity of 70%; and

3) Verifying a system flow rate of 7600 cfm to 9800 cfm during system operation when tested in accordance with ANSI N510-1980.

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F) and a relative humidity of 70%:

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 6.25 inches Water Gauge while operating the system at a flow rate of 7600 cfm to 9800 cfm,

2) Verifying that the system starts on a Safety Injection test signal, and

3) Verifying that the heaters dissipate 50 ±5 kW when tested in accordance with ANSI N510-1980.

---

* ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system 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 7600 cfm to 9800 cfm; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass 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 7600 cfm to 9800 cfm.

SECONDARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.6.6.2 Secondary Containment shall be OPERABLE.

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

ACTION:

With Secondary Containment inoperable, restore Secondary Containment to OPERABLE status 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 REQUIREMENT

4.6.6.2.1 OPERABILITY of Secondary Containment shall be demonstrated at the frequency specified in the Surveillance Frequency Control Program by verifying that each door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

4.6.6.2.2 At the frequency specified in the Surveillance Frequency Control Program, verify each Supplementary Leak Collection and Release System produces a negative pressure of greater than or equal to 0.4 inch water gauge in the Auxiliary Building at 24'-6" elevation within 120 seconds after a start signal.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.3 The structural integrity of the Secondary Containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.6.3.

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

ACTION:

With the structural integrity of the Secondary Containment 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 REQUIREMENT

4.6.6.3 The structural integrity of the Secondary Containment shall be determined at the frequency specified in the Containment Leakage Rate Testing Program, by a visual inspection of the exposed accessible interior and exterior surfaces of the Secondary Containment and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the Secondary Containment detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days.

## 3/4.7 PLANT SYSTEMS

### 3/4.7.1 TURBINE CYCLE

### SAFETY VALVES

### LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves (MSSVs) shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

Separate Condition entry is allowed for each MSSV.

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

a. With one or more steam generators (SGs) with one MSSV inoperable, and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels, within 4 hours reduce THERMAL POWER to less than or equal to 59% RATED THERMAL POWER (RTP); 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 SGs with two or more MSSVs inoperable, within 4 hours reduce THERMAL POWER to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs, and reduce the Power Range Neutron Flux High setpoint to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for number of OPERABLE MSSVs within the next 32 hours*; otherwise, 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 SGs with one MSSV inoperable and the MTC positive at any power level, within 4 hours reduce THERMAL POWER to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs and reduce the Power Range Neutron Flux High setpoint to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for number of OPERABLE MSSVs within the next 32 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

---

* Applicable only in MODE 1.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

d. With one or more SGs with four or more MSSVs inoperable, 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 Surveillance Requirements other than those required by Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

**TABLE 3.7-1**
**OPERABLE MSSVS VERSUS MAXIMUM ALLOWABLE POWER**

| NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR | MAXIMUM ALLOWABLE POWER (PERCENT OF RATED THERMAL POWER) |
|---|---|
| 4 | 59 |
| 3 | 41 |
| 2 | 24 |

**TABLE 3.7-2**

DELETED

## TABLE 3.7-3

## STEAM LINE SAFETY VALVES PER LOOP

| VALVE NUMBER | LIFT SETTING* (±3%)** | ORIFICE SIZE |
|---|---|---|
| **LOOP 1** | | |
| RV22A | 1185 psig | 16.0 square inches |
| RV23A | 1195 psig | 16.0 square inches |
| RV24A | 1205 psig | 16.0 square inches |
| RV25A | 1215 psig | 16.0 square inches |
| RV26A | 1225 psig | 16.0 square inches |
| **LOOP 2** | | |
| RV22B | 1185 psig | 16.0 square inches |
| RV23B | 1195 psig | 16.0 square inches |
| RV24B | 1205 psig | 16.0 square inches |
| RV25B | 1215 psig | 16.0 square inches |
| RV26B | 1225 psig | 16.0 square inches |
| **LOOP 3** | | |
| RV22C | 1185 psig | 16.0 square inches |
| RV23C | 1195 psig | 16.0 square inches |
| RV24C | 1205 psig | 16.0 square inches |
| RV25C | 1215 psig | 16.0 square inches |
| RV26C | 1225 psig | 16.0 square inches |
| **LOOP 4** | | |
| RV22D | 1185 psig | 16.0 square inches |
| RV23D | 1195 psig | 16.0 square inches |
| RV24D | 1205 psig | 16.0 square inches |
| RV25D | 1215 psig | 16.0 square inches |
| RV26D | 1225 psig | 16.0 square inches |

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

**The lift setting shall be within ± 1% following main steam line Code safety valve testing.

AUXILIARY FEEDWATER SYSTEM

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 busses, 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

– – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – – –

| Inoperable Equipment | Required ACTION |
|---|---|
| a. Turbine-driven auxiliary feedwater pump due to one of the two required steam supplies being inoperable. | a. Restore affected equipment to OPERABLE status within 7 days. If these ACTIONS are not met, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. |
| b. – – – – NOTE – – – – – Only applicable if MODE 2 has not been entered following REFUELING. – – – – – – – – – – – – – – – One turbine-driven auxiliary feedwater pump in MODE 3 following REFUELING. | b. Restore affected equipment to OPERABLE status within 7 days. If these ACTIONS are not met, be in at least HOT SHUTDOWN within the following 12 hours. |
| c. One auxiliary feedwater pump in MODE 1, 2, or 3 for reasons other than a. or b. above. | c. Restore the auxiliary feedwater pump to OPERABLE status within 72 hours. If these ACTIONS are not met, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. |
| d. Two auxiliary feedwater pumps in MODE 1, 2, or 3. | d. Be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 12 hours. |

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

| Inoperable Equipment | Required ACTION |
|---|---|
| e. Three auxiliary feedwater pumps in MODE 1, 2, or 3. | e. ---------- NOTE ---------- LCO 3.0.3 and all other LCO required ACTIONS requiring MODE changes are suspended until one AFW pump is restored to OPERABLE status. -------------------- Immediately initiate ACTION to restore one auxiliary feedwater pump to OPERABLE status. |

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:

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

Auxiliary feedwater pumps may be considered OPERABLE during alignment and operation for steam generator level control, if they are capable of being manually realigned to the auxiliary feedwater mode of operation.

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

Verifying each auxiliary feedwater manual, power operated, and automatic valve in each water flow path and in each required steam supply flow path to the steam turbine driven auxiliary feedwater pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.

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

b. Not required to be performed for the steam turbine driven auxiliary feedwater pump until 24 hours after reaching 800 psig in the steam generators.

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

Verify the developed head of each auxiliary feedwater pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable to the steam turbine driven auxiliary feedwater pump for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

c. At the frequency specified in the Surveillance Frequency Control Program by verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal. For the steam turbine-driven auxiliary feedwater pump, the provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying flow to each steam generator.

## DEMINERALIZED WATER STORAGE TANK

### LIMITING CONDITION FOR OPERATION

3.7.1.3 The demineralized water storage tank (DWST) shall be OPERABLE with a water volume of at least 334,000 gallons.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the DWST inoperable, within 4 hours either:

a. Restore the DWST 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 condensate storage tank (CST) as a backup supply to the auxiliary feedwater pumps and restore the DWST 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 DWST shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying the water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The CST shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying that the combined volume of both the DWST and CST is at least 384,000 gallons of water whenever the CST and DWST are 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**

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY |
|---|---|
| 1. Gross Radioactivity Determination | 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. |

## MAIN STEAM LINE ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.7.1.5 Four main steam line isolation valves (MSIVs) shall be OPERABLE.

APPLICABILITY: MODE 1

MODES 2, 3, and 4, except when all MSIVs are closed and deactivated.

ACTION:

MODE 1:

With one MSIV inoperable, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 8 hours; otherwise be in MODE 2 within the next 6 hours.

MODES 2, 3, and 4:

With one or more MSIVs inoperable, subsequent operation in MODE 2, or 3, or 4 may proceed provided the inoperable isolation valve(s) is (are) closed* within 8 hours and verified closed once per 7 days. Otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Separate condition entry is allowed for each MSIV.

### SURVEILLANCE REQUIREMENTS

4.7.1.5.1 DELETED

4.7.1.5.2 Each MSIV shall be demonstrated OPERABLE, pursuant to Specification 4.0.5, by verifying full closure within 10 seconds (120 seconds for MODE 4 only) on an actual or simulated actuation signal. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4 or MODE 3.

*The MSIVs may be opened to perform Surveillance Requirement 4.7.1.5.2 when Reactor Coolant System temperature is greater than or equal to 320°F.

## STEAM GENERATOR ATMOSPHERIC RELIEF BYPASS LINES

### LIMITING CONDITION FOR OPERATION

3.7.1.6 Each steam generator atmospheric relief bypass valve (SGARBV) line shall be OPERABLE, with the associated main steam atmospheric relief isolation (block) valve in the open position.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS:

a. With one required SGARBV line inoperable, restore required SGARBV line to OPERABLE status within 7 days or be in at least MODE 3 within the next 6 hours and be in MODE 4 without reliance upon steam generator for heat removal within the next 18 hours.

b. With two or more required SGARBV lines inoperable, restore all but one required SGARBV line to OPERABLE status within 24 hours or be in at least MODE 3 within the next 6 hours and be in MODE 4 without reliance upon steam generator for heat removal within the next 18 hours.

### SURVEILLANCE REQUIREMENTS

4.7.1.6.1 Verify one complete cycle of each SGARBV at the frequency specified in the Surveillance Frequency Control Program.

4.7.1.6.2 Verify one complete cycle of each main steam atmospheric relief isolation (block) valve at the frequency specified in the Surveillance Frequency Control Program.

THIS PAGE INTENTIONALLY LEFT BLANK

## 3/4.7.3 REACTOR PLANT COMPONENT COOLING WATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent reactor plant component cooling water safety loops shall be OPERABLE.

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

ACTION:

With only one reactor plant component cooling water safety loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.3 At least two reactor plant component cooling water safety 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 actuates to its correct position on its associated Engineered Safety Feature actuation signal, and

   2) Each Component Cooling Water System pump starts automatically on an SIS test signal.

## 3/4.7.4 SERVICE WATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent service water loops shall be OPERABLE.

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

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.4 At least two 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 actuates to its correct position on its associated Engineered Safety Feature actuation signal, and

2) Each Service Water System pump starts automatically on an SIS test signal.

## 3/4.7.5 ULTIMATE HEAT SINK

### LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with a water temperature of less than or equal to 80°F.

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

ACTION:

With the UHS water temperature greater than 80°F, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.5 The UHS shall be determined OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying the water temperature to be within limits.

b. At least once per 6 hours by verifying the water temperature to be within limits when the water temperature exceeds 75°F.

THIS PAGE INTENTIONALLY LEFT BLANK

## 3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Control Room Emergency Air Filtration Systems shall be OPERABLE.*

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

ACTION:

MODES 1, 2, 3 and 4:

a. With one Control Room Emergency Air Filtration System inoperable, except as specified in ACTION c., 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 both Control Room Emergency Air Filtration Systems inoperable, except as specified in ACTION c., restore at least one inoperable system to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With one or more Control Room Emergency Air Filtration Systems inoperable due to an inoperable CRE boundary, perform the following:

   1. Immediately initiate action to implement mitigating actions, and

   2. Verify, within 24 hours, mitigating actions ensure CRE occupant exposures to radiological and chemical hazards will not exceed limits, and mitigating actions are taken for exposure to smoke hazards, and

   3. Restore CRE boundary to OPERABLE status within 90 days.

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

During movement of recently irradiated fuel assemblies:

d. With one Control Room Emergency Air Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE Control Room Emergency Air Filtration System in the emergency mode of operation, or immediately suspend the movement of recently irradiated fuel assemblies.

---

* The Control Room Envelope (CRE) boundary may be opened intermittently under administrative control.

ACTION: (Continued)

e. With both Control Room Emergency Air Filtration Systems inoperable, or with the OPERABLE Control Room Emergency Air Filtration System required to be in the emergency mode by ACTION d. not capable of being powered by an OPERABLE emergency power source, or with one or more Control Room Emergency Air Filtration System Trains inoperable due to an inoperable CRE boundary, immediately suspend the movement of recently irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Control Room Emergency Air Filtration System shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 95°F;

b. 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 a system flow rate of 1,120 cfm ± 20% and that the system operates for at least 15 continuous minutes with the heaters operating;

c. At the frequency specified in the Surveillance Frequency Control Program and following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the 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, Revisions 2, March 1978,* and the system flow rate is 1,120 cfm ± 20%;

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 54 ft/min; and

3) Verifying a system flow rate of 1,120 cfm ± 20% during system operation when tested in accordance with ANSI N510-1980.

SURVEILLANCE REQUIREMENTS (Continued)

d. 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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), and a relative humidity of 70%, and a face velocity of 54 ft/min.

e. 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 6.75 inches Water Gauge while operating the system at a flow rate of 1,120 cfm ± 20%;

   2) Deleted

   3) Verifying that the heaters dissipate 9.4 ±1 kW when tested in accordance with ANSI N510-1980.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system 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 1120 cfm ± 20%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass 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 1120 cfm ± 20%.

h. By performance of CRE unfiltered air inleakage testing in accordance with the CRE Habitability Program at a frequency in accordance with the CRE Habitability Program.

---

* ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

PLANT SYSTEMS

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 Two independent Auxiliary Building Filter Systems shall be OPERABLE.

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

ACTION:

With one Auxiliary Building Filter 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. In addition, comply with the ACTION requirements of Specification 3.6.6.1.

SURVEILLANCE REQUIREMENTS

4.7.9 Each Auxiliary Building Filter 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 a system flow rate of 30,000 cfm ±10% and that the system operates for at least 15 continuous minutes with the heaters operating;

b. At the frequency specified in the Surveillance Frequency Control Program and following 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 system flow rate is 30,000 cfm ±10%;

    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,* shows the methyl

iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 52 ft/min; and

3) Verifying a system flow rate of 30,000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 52 ft/min;

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 6.8 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ±10%,

2) Verifying that the system starts on a Safety Injection test signal, and

3) Verifying that the heaters dissipate 180 ±18 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system 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 30,000 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 and bypass 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 30,000 cfm ±10%.

* ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

## 3/4.7.10 SNUBBERS

### LIMITING CONDITION FOR OPERATION

3.7.10 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 Specification 4.7.10. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

### SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Examination, Testing, and Service Life Monitoring Program Plan.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

PLANT SYSTEMS

3/4.7.14 DELETED

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent 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 volume of 278 gallons of fuel,

   2. A separate Fuel Storage System containing a minimum volume of 32,760 gallons of fuel,

   3. A separate fuel transfer pump,

   4. Lubricating oil storage containing a minimum total volume of 280 gallons of lubricating oil, and

   5. Capability to transfer lubricating oil from storage to the diesel generator unit.

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

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

LCO 3.0.4.b is not applicable to diesel generators

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

ACTION:

| Inoperable Equipment | Required ACTION |
|---|---|
| a. One offsite circuit | a.1 Perform Surveillance Requirement 4.8.1.1.1.a for remaining offsite circuit within 1 hour prior to or after entering this condition, and at least once per 8 hours thereafter.<br><br>AND<br><br>a.2 Restore the inoperable offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. |
| b. One diesel generator | b.1 Perform Surveillance Requirement 4.8.1.1.1.a for the offsite circuits within 1 hour prior to or after entering this condition, and at least once per 8 hours thereafter.<br><br>AND<br><br>b.2 Demonstrate OPERABLE diesel generator is not inoperable due to common cause failure within 24 hours or perform Surveillance Requirement 4.8.1.1.2.a.5 for the OPERABLE diesel generator within 24 hours.<br><br>AND |

ACTION (continued)

| Inoperable Equipment | Required ACTION |
|---|---|
| b. One diesel generator | b.3 Verify all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are OPERABLE, and the steam-driven auxiliary feedwater pump is OPERABLE (MODES 1, 2, and 3 only). If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.<br>AND<br>b.4 (Applicable only if the 14 day allowed outage time specified in ACTION Statement b.5 is to be used). Verify the required Millstone Unit No. 2 diesel generator(s) is/are OPERABLE and the Millstone Unit No. 3 SBO diesel generator is available within 1 hour prior to or after entering this condition, and at least once per 24 hours thereafter. Restore any inoperable required Millstone Unit No. 2 diesel generator to OPERABLE status and/or Millstone Unit No. 3 SBO diesel generator to available status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.<br>AND<br>b.5 Restore the inoperable diesel generator to OPERABLE status within 72 hours (within 14 days if ACTION Statement b.4 is met) or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. |
| c. One offsite circuit<br>AND<br>One diesel generator | c.1 Perform Surveillance Requirement 4.8.1.1.1.a for remaining offsite circuit within 1 hour and at least once per 8 hours thereafter.<br>AND<br>c.2 Demonstrate OPERABLE diesel generator is not inoperable due to common cause failure within 8 hours or perform Surveillance Requirement 4.8.1.1.2.a.5 for the OPERABLE diesel generator within 8 hours.<br>AND |

ACTION (continued)

| Inoperable Equipment | Required ACTION |
|---|---|
| c. One offsite circuit<br>AND<br>One diesel generator | c.3 Verify all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are OPERABLE, and the steam-driven auxiliary feedwater pump is OPERABLE (MODES 1, 2, and 3 only). If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.<br>AND<br>c.4 Restore one inoperable A.C. source to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.<br>AND<br>c.5 Restore remaining inoperable A.C. source to OPERABLE status following the time requirements of ACTION Statements a. or b. above based on the initial loss of the remaining inoperable A.C. source. |
| d. Two offsite circuits | d.1 Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours.<br>AND<br>d.2 Following restoration of one offsite source, restore remaining inoperable offsite source to OPERABLE status following the time requirements of ACTION Statement a. above based on the initial loss of the remaining inoperable offsite source. |
| e. Two diesel generators | e.1 Perform Surveillance Requirement 4.8.1.1.1.a for the offsite circuits within 1 hour and at least once per 8 hours thereafter.<br>AND |

ACTION (continued)

| Inoperable Equipment | Required ACTION |
|---|---|
| e. Two diesel generators | e.2 Restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.<br><br>AND<br><br>e.3 Following restoration of one diesel generator, restore remaining inoperable diesel generator to OPERABLE status following the time requirements of ACTION Statement b. above based on the initial loss of the remaining inoperable diesel generator. |

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each 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 fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4) Verifying the lubricating oil inventory in storage,

5) Verifying the diesel starts from standby conditions and achieves generator voltage and frequency at 4160 ± 420 volts and 60 ± 0.8 Hz. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

* All planned starts for the purpose of these surveillances may be preceded by an engine prelube period.

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

6) Verifying the generator is synchronized and gradually loaded in accordance with the manufacturer's recommendations between 4800-5000 kW* and operates with a load between 4800-5000 kW* for at least 60 minutes, and

7) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

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

1) Verifying that the diesel generator starts from standby conditions and attains generator voltage and frequency of 4160 ± 420 volts and 60 ± 0.8 Hz within 11 seconds after the start signal.

2) Verifying the generator is synchronized to the associated emergency bus, loaded between 4800-5000 kW* in accordance with the manufacturer's recommendations, and operate with a load between 4800-5000 kW* for at least 60 minutes.

The diesel generator shall be started for this test using one of the signals in Surveillance Requirement 4.8.1.1.2.a.5. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

c. 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 or equal to 1 hour by checking for and removing accumulated water from the day tank;

d. At the frequency specified in the Surveillance Frequency Control Program by checking for and removing accumulated water from the fuel oil storage tanks;

e. By sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

---

* The operating band is meant as guidance to avoid routine overloading of the diesel. Momentary transients outside the load range shall not invalidate the test.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

c) A flash point equal to or greater than 125°F; and

d) Water and sediment less than 0.05 percent by volume when tested in accordance with ASTM-D1796-83.

2. By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that: (1) the cetane index shall be determined in accordance with ASTM-D976 (this test is an appropriate approximation for cetane number as stated in ASTM-D975-81 [Note E]), and (2) the analysis for sulfur may be performed in accordance with ASTM-D1552-79, ASTM-D2622-82 or ASTM-D4294-83.

f. At the frequency specified in the Surveillance Frequency Control Program by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

g. At the frequency specified in the Surveillance Frequency Control Program, during shutdown, by:

1) DELETED

2) Verifying the generator capability to reject a load of greater than or equal to 595 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 3 Hz;

3) Verifying the generator capability to reject a load of 4986 kW without tripping. The generator voltage shall not exceed 5000 volts during and 4784 volts following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts from standby conditions on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 0.8 Hz during this test.

SURVEILLANCE REQUIREMENTS (Continued)

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts from standby conditions on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 0.8 Hz within 11 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts from standby conditions on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$ 420 volts and 60 $\pm$ 0.8 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed, lube oil pressure low (2 of 3 logic) and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) DELETED

8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 5335 kW;

9) 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) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) DELETED

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval; and

13) DELETED

h. At the frequency specified in the Surveillance Frequency Control Program by starting both diesel generators simultaneously from standby conditions, during shutdown, and verifying that both diesel generators achieve generator voltage and frequency at 4160 ± 420 volts and 60 ± 0.8 Hz in less than or equal to 11 seconds; and

i. At the frequency specified in the Surveillance Frequency Control Program by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution.

j. At the frequency specified in the Surveillance Frequency Control Program by verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded between 5400-5500kW* and during the remaining 22 hours of this test, the diesel generator shall be loaded between 4800-5000kW*. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 0.8 Hz within 11 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test.** Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2.a.5) excluding the requirement to start the diesel from standby conditions.***

k. At the frequency specified in the Surveillance Frequency Control Program by verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines.

l. At the frequency specified in the Surveillance Frequency Control Program by verifying that the following diesel generator lockout features prevent diesel generator starting:

1) Engine overspeed,

2) Lube oil pressure low (2 of 3 logic),

3) Generator differential, and

4) Emergency stop.

---

* The operating band is meant as guidance to avoid routine overloading of the diesel. Momentary transients outside the load range shall not invalidate the test.

** Diesel generator loadings may include gradual loading as recommended by the manufacturer.

*** If Surveillance Requirement 4.8.1.1.2.a.5) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated between 4800-5000 kW for 2 hours or until operating temperature has stabilized.

This page intentionally left blank.

A. C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A. C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

1) A day tank containing a minimum volume of 278 gallons of fuel,

2) A fuel storage system containing a minimum volume of 32,760 gallons of fuel,

3) A fuel transfer pump,

4) Lubricating oil storage containing a minimum total volume of 280 gallons of lubricating oil, and

5) Capability to transfer lubricating oil from storage to the diesel generator unit.

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 additions that could result in loss of required SDM or boron concentration, movement of irradiated fuel, crane operation with loads over the fuel storage pool, or operation with a potential for draining the reactor vessel; initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENT

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, 4.8.1.1.2 (except for Specifications 4.8.1.1.2.a.6 and 4.8.1.1.2.b.2).

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery Bank 301A-1, and an associated full capacity charger,

b. 125-volt Battery Bank 301A-2, and an associated full capacity charger,

c. 125-volt Battery Bank 301B-1 and an associated full capacity charger, and

d. 125-volt Battery Bank 301B-2 and an associated full capacity charger.

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

ACTION:

a. With either Battery Bank 301A-1 or 301B-1, and/or one of the required full capacity chargers inoperable, restore the inoperable battery bank and/or full capacity charger 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.

b. With either Battery Bank 301A-2 or 301B-2 inoperable, and/or one of the required full capacity chargers inoperable, restore the inoperable battery bank and/or full capacity charger to OPERABLE status 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

4.8.2.1 Each 125-volt battery bank 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 4.8-2a 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 4.8-2a meet the Category B limits,

2) There is no visible corrosion at either terminals or connectors, or the connection resistance of each cell-to-cell and terminal connection is less than $150 \times 10^{-6}$ ohm and total battery resistance is less than $3700 \times 10^{-6}$ ohm, and

3) The average electrolyte temperature of six connected cells is above 60°F.

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 $150 \times 10^{-6}$ ohm and total battery resistance is less than $3700 \times 10^{-6}$ ohm, and

4) Each battery charger will supply at least the amperage indicated in Table 4.8-2b at greater than or equal to 132 volts for at least 24 hours.

d. At the frequency specified in the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. At the frequency specified in the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, 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.

# TABLE 4.8-2a

## 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 | ≥ 2.13 volts | ≥ 2.13 volts[6] | > 2.07 volts |
| Specific Gravity[4] | ≥ 1.200[5] | ≥ 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 ≥ 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.

## TABLE 4.8-2b

## BATTERY CHARGER CAPACITY

| CHARGER | AMPERAGE |
|---|---|
| 301A-1 | 200 |
| 301A-2 | 50 |
| 301A-3 | 200 |
| 301B-1 | 200 |
| 301B-2 | 50 |
| 301B-3 | 200 |

D. C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one train (A or B) of batteries and their associated full capacity chargers shall be OPERABLE:

a. Train - "A" consisting of:

1) Battery Bank 301A-1 and a full capacity battery charger, and

2) Battery Bank 301A-2 and a full capacity battery charger.

OR

b. Train - "B" consisting of:

1) Battery Bank 301B-1 and a full capacity battery charger, and

2) Battery Bank 301B-2 and a full capacity battery charger.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required train inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity additions that could result in loss of required SDM or boron concentration, movement of recently irradiated fuel assemblies; crane operation with loads over the fuel storage pool, or operation with a potential for draining the reactor vessel; initiate corrective action to restore the required train to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required train 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 busses shall be OPERABLE in the specified manner:

a. Train A A.C. Emergency Busses consisting of:

1) 4160-Volt Emergency Bus #34C, and

2) 480-Volt Emergency Bus #32R, 32S, 32T, and 32Y.

b. Train B A.C. Emergency Busses consisting of:

1) 4160-Volt Emergency Bus #34D, and

2) 480-Volt Emergency Bus #32U, 32V, 32W, and 32X.

c. 120-Volt A.C. Vital Bus #VIAC-1 energized from its associated inverter connected to D.C. Bus #301A-1*,

d. 120-Volt A.C. Vital Bus #VIAC-2 energized from its associated inverter connected to D.C. Bus #301B-1*,

e. 120-Volt A.C. Vital Bus #VIAC-3 energized from its associated inverter connected to D.C. Bus #301A-2*,

f. 120-Volt A.C. Vital Bus #VIAC-4 energized from its associated inverter connected to D.C. Bus #301B-2*,

g. 125-Volt D.C. Bus #301A-1 energized from Battery Bank #301A-1,

h. 125-Volt D.C. Bus #301A-2 energized from Battery Bank #301A-2,

i. 125-Volt D.C. Bus #301B-1 energized from Battery Bank #301B-1, and

j. 125-Volt D.C. Bus #301B-2 energized from Battery Bank #301B-2.

---

* Two inverters may be disconnected from their D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

## LIMITING CONDITION FOR OPERATION

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

ACTION:

a. With one of the required trains of A.C. emergency busses not OPERABLE, restore the inoperable train to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

## SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined OPERABLE in the specified manner at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, one train (A or B) of the following electrical busses shall be OPERABLE:

a. Train - "A" consisting of:

1) One 4160 volt AC Emergency Bus #34C, and

2) Four 480 volt AC Emergency Busses #32R, #32S, #32T, #32Y, and

3) Two 120 volt AC Vital Busses consisting of:

a) Bus #VIAC-1 energized from Inverter #INV-1 connected to DC Bus #301A-1, and

b) Bus #VIAC-3 energized from Inverter #INV-3 connected to DC Bus #301A-2, and

4) Two 125 volt DC Busses consisting of:

a) Bus #301A-1 energized from Battery Bank #301A-1, and

b) Bus #301A-2 energized from Battery Bank #301A-2.

OR

b. Train - "B" consisting of

1) One 4160 volt AC Emergency Bus #34D, and

2) Four 480 volt AC Emergency Busses #32U, #32V, #32W, #32X, and

3) Two 120 volt AC Vital Busses consisting of:

a) Bus #VIAC-2 energized from Inverter #INV-2 connected to DC Bus #301B-1, and

b) Bus #VIAC-4 energized from Inverter #INV-4 connected to DC Bus #301B-2, and

LIMITING CONDITION FOR OPERATION (Continued)

4) Two 125 volt DC Busses consisting of:

   a) Bus #301B-1 energized from Battery Bank #301B-1, and

   b) Bus #301B-2 energized from Battery Bank #301B-2.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity additions that could result in loss of required SDM or boron concentration, movement of recently irradiated fuel assemblies, crane operation with loads over the fuel storage pool, or operations with a potential for draining the reactor vessel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses 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 busses.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

## 3/4.9 REFUELING OPERATIONS

### 3/4.9.1 BORON CONCENTRATION

### LIMITING CONDITION FOR OPERATION

3.9.1.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A $K_{eff}$ of 0.95 or less, or

b. A boron concentration of greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

Additionally, the CVCS valves of Specification 4.1.1.2.2 shall be closed and secured in position.

APPLICABILITY: MODE 6.*

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and positive reactivity additions and initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or its equivalent until $K_{eff}$ is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to the limit specified in the COLR, whichever is the more restrictive.

b. With any of the CVCS valves of Specification 4.1.1.2.2 not closed** and secured in position, immediately close and secure the valves.

### SURVEILLANCE REQUIREMENTS

4.9.1.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.1.2 The boron concentration of the Reactor Coolant System and the refueling cavity shall be determined by chemical analysis at the frequency specified in the Surveillance Frequency Control Program.

4.9.1.1.3 The CVCS valves of Specification 4.1.1.2.2 shall be verified closed and locked at the frequency specified in the Surveillance Frequency Control Program.

---

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

** Except those opened under administrative control.

REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be greater than or equal to 2600 ppm.

APPLICABILITY:

Whenever fuel assemblies are in the spent fuel pool.

ACTION:

a. With the boron concentration less than 2600 ppm, initiate action to bring the boron concentration in the fuel pool to at least 2600 ppm within 72 hours, and

b. With the boron concentration less than 2600 ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to 2600 ppm at the frequency specified in the Surveillance Frequency Control Program.

## 3/4.9.2 INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.9.2 Two Source Range Neutron Flux Monitors shall be OPERABLE with continuous visual indication in the control room, and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable immediately suspend all operations involving CORE ALTERATIONS and operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1.

b. With both of the above required monitors inoperable determine the boron concentration of the Reactor Coolant System within 4 hours and at least once per 12 hours thereafter.

### SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK and verification of audible counts at the frequency specified in the Surveillance Frequency Control Program,

b. A CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program.*

* Neutron detectors are excluded from CHANNEL CALIBRATION.

## 3/4.9.3 DECAY TIME

### LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

### SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

## 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 access hatch shall be either:

1. closed and held in place by a minimum of four bolts, or

2. open under administrative control * and capable of being closed and held in place by a minimum of four bolts,

b. A personnel access hatch shall be either:

1. closed by one personnel access hatch door, or

2. capable of being closed by an OPERABLE personnel access hatch door, under administrative control,* and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed under administrative control.*

APPLICABILITY: During movement of fuel within the containment building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of fuel in the containment building.

### SURVEILLANCE REQUIREMENTS

4.9.4.a Verify each required containment penetrations is in the required status at the frequency specified in the Surveillance Frequency Control Program.

4.9.4.b DELETED

---

* Administrative controls shall ensure that appropriate personnel are aware that the equipment access hatch penetration, personnel access hatch doors and/or other containment penetrations are open, and that a specific individual(s) is designated and available to close the equipment access hatch penetration, a personnel access hatch door and/or other containment penetrations within 30 minutes if a fuel handling accident occurs. Any obstructions (e.g. cables and hoses) that could prevent closure of the equipment access hatch penetration, a personnel access hatch door and/or other containment penetrations must be capable of being quickly removed.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

## 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, 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 or in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1 and suspend loading irradiated fuel assemblies in the core 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 2800 gpm at the frequency specified in the Surveillance Frequency Control Program.

4.9.8.1.2 Locations susceptible to gas accumulation in the required RHR trains shall be verified to be sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

* The RHR loop may be removed from operation for up to 1 hour per 8-hour period, provided no operations are permitted that could cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1.

## LOW WATER LEVEL

### 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, 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 operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1 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 2800 gpm at the frequency specified in the Surveillance Frequency Control Program.

4.9.8.2.2 Locations susceptible to gas accumulation in the required RHR trains shall be verified to be sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

* The RHR loop may be removed from operation for up to 1 hour per 8-hour period, provided no operations are permitted that could cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1.

THIS PAGE INTENTIONALLY LEFT BLANK

## 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: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

### 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 - STORAGE POOL

### LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas 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 The water level in the storage pool shall be determined to be at least its minimum required depth at the frequency specified in the Surveillance Frequency Control Program when irradiated fuel assemblies are in the fuel storage pool.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

## 3/4.9.13 SPENT FUEL POOL - STORAGE

### LIMITING CONDITION FOR OPERATION

3.9.13 The spent fuel storage requirements necessary to maintain $K_{eff}$ within limits shall be met.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION:

a. For a fuel assembly stored in Region 1A - initiate immediate action to move any assembly which does not meet Surveillance Requirement 4.9.13.1.1 to Region 1B.

b. For a fuel assembly stored in Region 2 that does not contain a Rod Cluster Control Assembly - initiate immediate action to move any assembly which does not meet the requirements of Figure 3.9-2 to a location for which that fuel assembly is allowed.

c. For a fuel assembly stored in Region 3 - initiate immediate action to move any assembly which does not meet the requirements of Figure 3.9-3 to a location for which that fuel assembly is allowed.

### SURVEILLANCE REQUIREMENTS

NOTE

The Region 1 Fuel Storage Loading Schematic (Figure 3.9-1) designates each storage location as either Region 1A or Region 1B.

Regarding fuel assemblies that contain a Rod Cluster Control Assembly for storage in Region 2 - if the enrichment and burnup of a given assembly is not in the "Acceptable" domain of Figure 3.9-2 (e.g., the assembly requires a Rod Cluster Control Assembly to be stored in Region 2), then the assembly must be located in an acceptable Region 1 storage location before its Rod Cluster Control Assembly can be inserted or removed.

Initial enrichment is the maximum initial planar volume averaged as-built U-235 enrichment in the assembly. If the assembly has axial blankets the lower enriched fuel is not credited in determining the enrichment. Also, fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates.

4.9.13.1.1. Ensure that all fuel assemblies to be placed into a Region 1A storage location, with an initial enrichment greater than 4.75 w/o U-235, have achieved a fuel burnup greater than or equal to 2.0 GWD/MTU or contain a minimum of twelve (12) Integral Fuel Burnable Absorber (IFBA) Rods by checking the fuel assembly's location, design, and burnup documentation. Fuel assemblies with an initial enrichment less than or equal to 4.75 w/o U-235 may be stored in Region 1A without restriction.

4.9.13.1.2. Ensure that all fuel assemblies to be placed in Region 1B are stored consistent with the Fuel Storage Loading Schematic specified in Figure 3.9-1 by checking the fuel assembly's storage location. All fuel assemblies with an initial enrichment less than or equal to 5.0 w/o U-235 may be stored in Region 1B without restriction.

4.9.13.1.3. Ensure that all fuel assemblies to be stored in Region 2 - that do not contain a Rod Cluster Control Assembly - are within the enrichment and burnup limits of Figure 3.9-2 by checking the fuel assembly's design and burnup documentation. A fuel assembly that contains a Rod Cluster Control Assembly may be stored in Region 2 without restriction.

4.9.13.1.4. Ensure that all fuel assemblies to be stored in Region 3 are within the enrichment, burnup, and decay time limits of Figure 3.9-3 by checking the fuel assembly's design, burnup, and decay time documentation.

THIS PAGE INTENTIONALLY LEFT BLANK

Figure 3.9-1 Region 1 Fuel Storage Loading Schematic

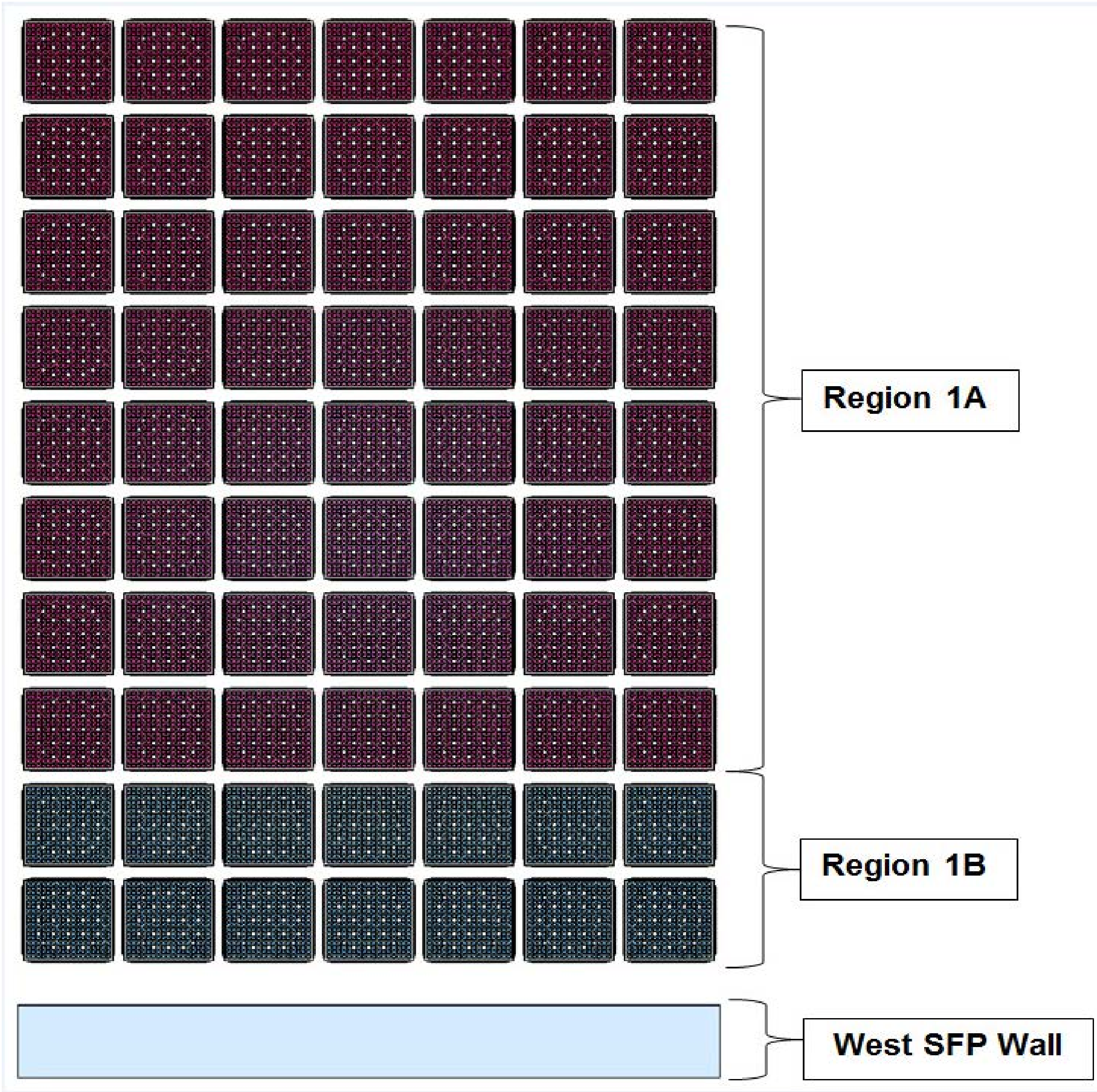


Figure 3.9-2 Minimum Fuel Assembly Burnup versus Initial Enrichment for Region 2 Storage Configuration
(Fuel Assemblies without Rod Cluster Control Assemblies)

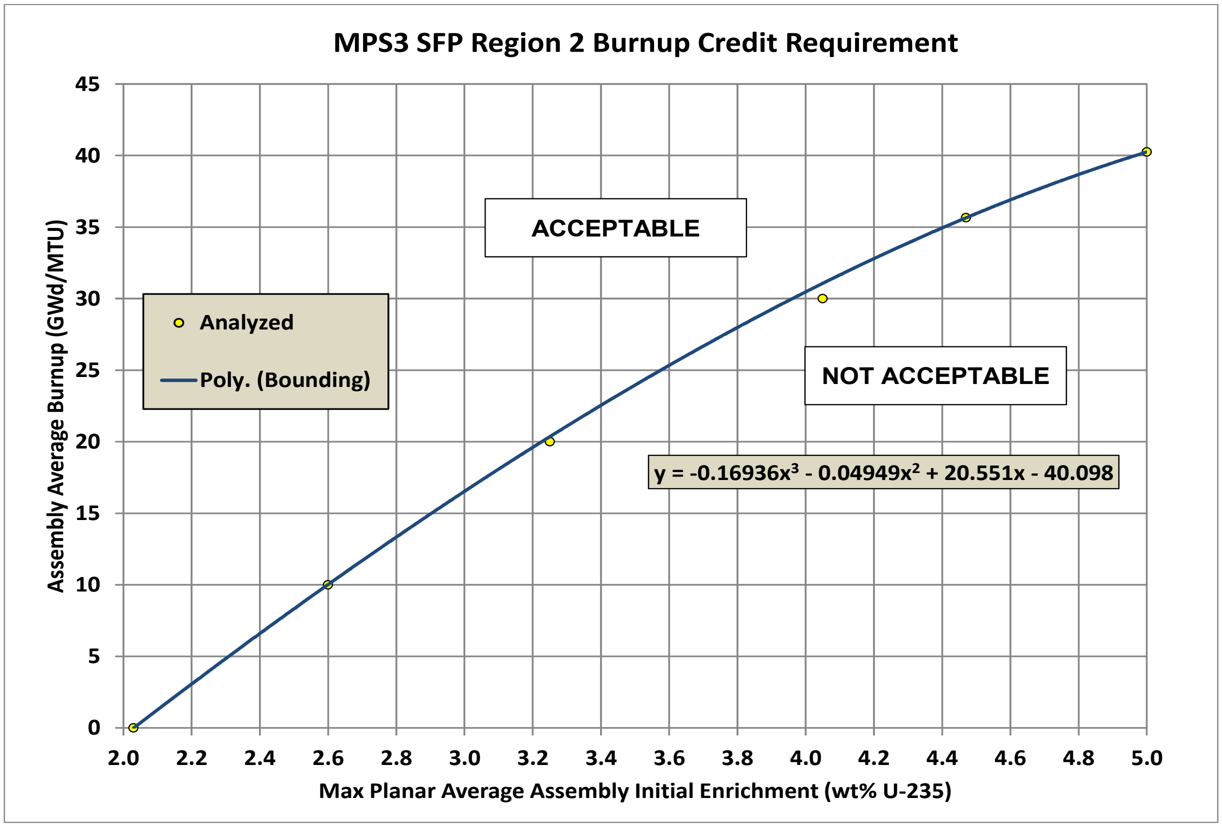
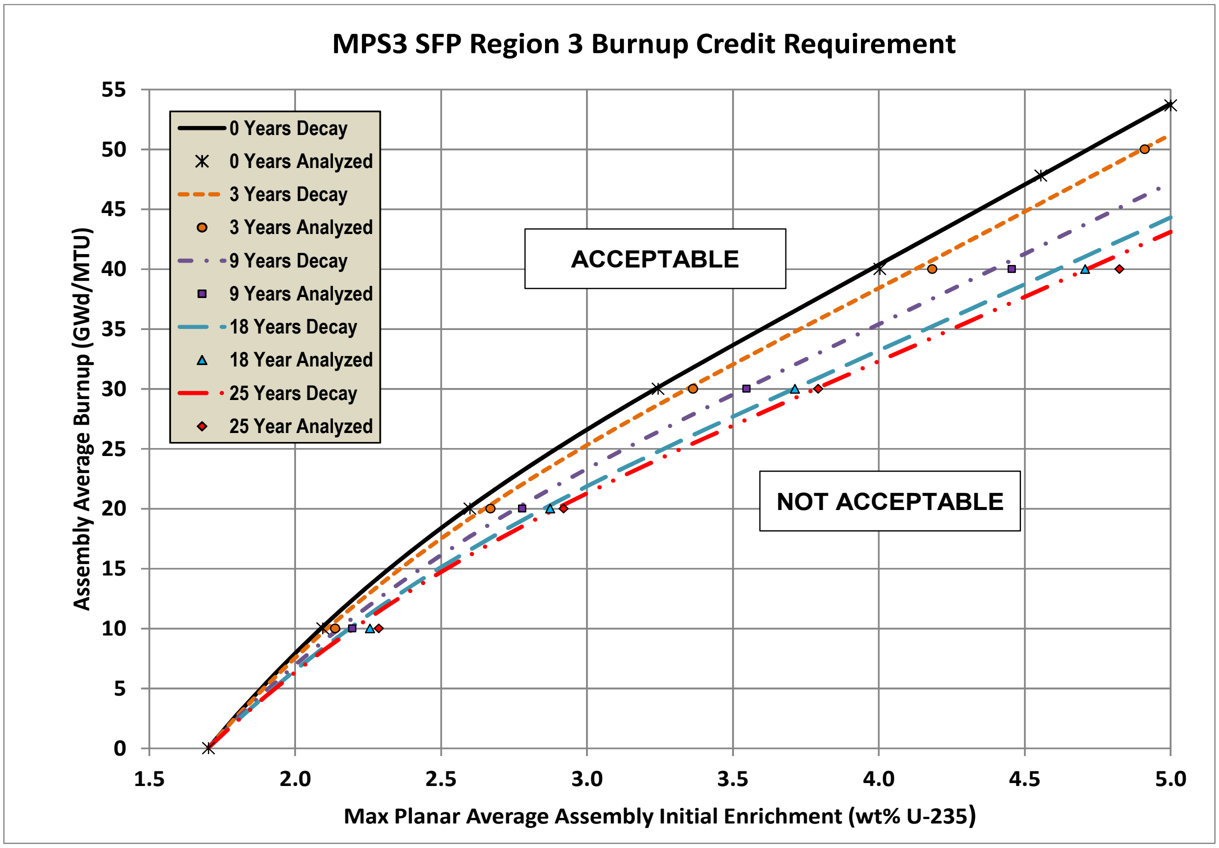


Figure 3.9-3 Minimum Fuel Assembly Burnup and Decay Time versus Initial Enrichment for Region 3 Storage Configuration




The burnup curve equations have the following polynomial format (bounding):

$$BU[GWD/MTU] = \alpha_4 * wt\%^4 + \alpha_3 * wt\%^3 + \alpha_2 * wt\%^2 + \alpha_1 * wt\%^1 + \alpha_0$$

**Burnup Credit Curve Polynomial Coefficients**

| Region | Decay Time (Years) | $\alpha_4$ | $\alpha_3$ | $\alpha_2$ | $\alpha_1$ | $\alpha_0$ |
|---|---|---|---|---|---|---|
| 3 | No Credit | -0.2459 | 4.208 | -26.80 | 88.70 | -92.00 |
| 3 | 3 Years | -0.2338 | 4.001 | -25.48 | 84.34 | -87.47 |
| 3 | 9 Years | -0.2153 | 3.684 | -23.46 | 77.66 | -80.54 |
| 3 | 18 Years | -0.2020 | 3.458 | -22.02 | 72.88 | -75.59 |
| 3 | 25 Years | -0.1964 | 3.361 | -21.40 | 70.84 | -73.47 |

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

## 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 control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

### SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length 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.

## 3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

### LIMITING CONDITION FOR OPERATION

3.10.2.1 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.1, 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 3.2.2.1 and 3.2.3.1 are maintained and determined at the frequencies specified in Specification 4.10.2.1.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2.1 or 3.2.3.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2.1 and 3.2.3.1, or

b. Be in HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.2.1.1 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.

4.10.2.1.2 The Surveillance Requirements of the below listed specifications shall be performed at the frequency specified in the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Specifications 4.2.2.1.2 and 4.2.2.1.3, and

b. Specification 4.2.3.1.2.

This page intentionally left blank.

## 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 within 12 hours 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 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 within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

THIS PAGE INTENTIONALLY LEFT BLANK

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

# SECTION 5.0

# DESIGN FEATURES

## 5.0 DESIGN FEATURES

### 5.1 SITE LOCATION

The Unit 3 Containment Building is located on the site at Millstone Point in Waterford, Connecticut. The nearest SITE BOUNDARY on land is 1719 feet northeast of the containment building wall (1627 feet northeast of the elevated stack), which is the minimum distance to the boundary of the exclusion area as described in 10 CFR 100.3. No part of the site that is closer than these distances shall be sold or leased except to Dominion Energy Nuclear Connecticut, Inc. or its corporate affiliates for use in conjunction with normal utility operations.

### 5.2 DELETED

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

## 5.3 REACTOR CORE

### FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies. Each fuel assembly shall consist of 264 zircaloy-4, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural uranium dioxide or a maximum nominal enrichment of 5.0 weight percent U-235 as fuel material. Limited substitutions of zircaloy-4, ZIRLO® or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assembly configurations shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or cycle-specific reload analyses to comply with all fuel safety design bases. Each fuel rod shall have a nominal active fuel length of 144 inches. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

### CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 61 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 95.3% hafnium and 4.5% natural zirconium or 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel.

## 5.4 DELETED

## 5.5 DELETED

## 5.6 FUEL STORAGE

### CRITICALITY

5.6.1 The New Fuel Storage Racks, a nominal 22.125 inch center to center distance, credit a fixed neutron absorber (BORAL) within the rack and are designed and shall be maintained with:

a. $K_{eff}$ less than or equal to 0.95 with the storage racks fully loaded with the highest reactivity fuel and flooded with potential moderators,

b. $K_{eff}$ less than or equal to 0.98 with the storage racks fully loaded with the highest reactivity fuel and optimum moderation of the racks.

The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a $K_{eff}$ less than 1.0 when flooded with unborated water, and $K_{eff}$ less than or equal to 0.95 with 600 ppm soluble boron in the spent fuel pool water. The storage rack regions are as follows:

a. Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack. Each Region 1 fuel storage rack contains two storage sub-regions - Region 1A and Region 1B:

(1) Region 1A - Fuel assemblies meeting one of the following three criteria may be stored in Region 1A storage locations:
i. initial enrichment less than or equal to 4.75 w/o U-235, or
ii. initial enrichment less than or equal to 5.0 w/o U-235 with a fuel burnup greater than or equal to 2.0 GWD/MTU, or
iii. initial enrichment less than or equal to 5.0 w/o U-235 that contain a minimum of 12 Integral Fuel Burnable Absorber (IFBA) rods.

(2) Region 1B - Fuel assemblies with an initial enrichment less than or equal to 5.0 w/o U-235 shall be stored per the Fuel Storage Loading Schematic shown in Figure 3.9-1 (the two rows against the spent fuel pool west wall are designated Region 1B).

b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack and either fuel burnup as shown in Figure 3.9-2 or takes credit for containing a Rod Cluster Control Assembly.

c. Region 3, a nominal 10.35 inch center to center distance, credits fuel burnup and decay time as shown in Figure 3.9-3. These racks contain Boraflex which is not credited.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

CAPACITY

5.6.3 The spent fuel storage pool contains 350 Region 1 storage locations, 673 Region 2 storage locations and 756 Region 3 storage locations, for a total of 1779 fuel storage locations. An additional Region 2 rack with 81 storage locations may be placed in the spent fuel pool, if needed. With this additional rack installed, the Region 2 storage capacity is 754 storage locations. The total storage capacity of the spent fuel pool is limited to no more than 1860 fuel assemblies.

5.7 DELETED

THIS PAGE INTENTIONALLY LEFT BLANK

# SECTION 6.0

# ADMINISTRATIVE CONTROLS

## 6.1 RESPONSIBILITY

6.1.1 The designated officer shall be responsible for overall operation of the Millstone Station Site and shall delegate in writing the succession to this responsibility. The designated manager shall be responsible for overall Unit safe operation and shall delegate in writing the succession to this responsibility.

6.1.2 The Shift Manager shall be responsible for the control room command function.

6.1.3 Unless otherwise defined, the technical specification titles for members of the staff are generic titles. Unit specific titles for the functions and responsibilities associated with these generic titles are identified in appropriate administrative documents.

## 6.2 ORGANIZATION

### 6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Program Topical Report.

b. The designated manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The designated officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

### 6.2.2 FACILITY STAFF

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

FACILITY STAFF (Continued)

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. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. A radiation protection technician* shall be on site when fuel is in the reactor;

e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

f. Deleted

g. Deleted

---

* The radiation protection 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.

## 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 |
| SM | 1 | 1 |
| SRO | 1 | None |
| RO | 2 | 1 |
| PEO | 2 | 1 |
| STA | 1* | None |

SM - Shift Manager with a Senior Operator license on Unit 3
SRO - Individual with a Senior Operator license on Unit 3
RO - Individual with an Operator license on Unit 3
PEO - Plant Equipment Operator (Non-licensed)
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 crewmember being late or absent.

During any absence of the Shift Manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Manager from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

*The STA position may be filled by an on-shift Senior Reactor Operator only if that Senior Reactor Operator meets the Shift Technical Advisor qualifications of the Commission Policy Statement on Engineering Expertise on Shift.

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 Shift Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions as specified in the Nuclear Facility Quality Assurance Program Description.

6.3.2 The operations manager or at least one operations middle manager shall hold a senior reactor operator license for Millstone Unit No. 3.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff that meets or exceeds the requirements as specified in the Quality Assurance Program and 10 CFR Part 55.59 shall be maintained.

6.4.2 Deleted.

6.5 DELETED.

PAGES 6-6 THROUGH 6-13 HAVE BEEN INTENTIONALLY DELETED.

6.6 Deleted.

6.7 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 applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto;

c. Refueling operations;

d. Surveillance activities of safety related equipment;

e. Not used.

f. Not used.

g. Fire Protection Program implementation;

h. Quality controls for effluent monitoring, using the guidance in Regulatory Guide 1.21, Rev. 1, June 1974; and

i. Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) implementation except for Section I.E, Radiological Environmental Monitoring.

6.8.2 a. The designated manager or designated officer or designated senior officer may designate specific procedures and programs, or classes of procedures and programs to be reviewed in accordance with the Quality Assurance Program Topical Report.

b. Procedures and programs listed in Specification 6.8.1, and changes thereto, shall be approved by the designated manager or designated officer or by cognizant managers or directors who are designated as the Approval Authority by designated manager or designated officer as specified in administrative procedures. The Approval Authority for each procedure and program or class of procedure and program shall be specified in administrative procedures.

c. Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed and approved in accordance with the Quality Assurance Program Topical Report, prior to implementation. Each procedure of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed and approved in accordance with the Quality Assurance Program Topical Report within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation spray, Safety Injection, charging portion of chemical and volume control, and hydrogen recombiners. The program shall include the following:

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 which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

PROCEDURES AND PROGRAMS (Continued)

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 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 the discharge of the condensate pumps 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

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. Deleted

e. Accident Monitoring Instrumentation

A program which will ensure the capability to monitor plant variables and systems operating status during and following an accident. This program shall include those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials (Category 2 and 3 instrumentation as defined in Regulatory Guide 1.97, Revision 2) and provide the following:

1) Preventive maintenance and periodic surveillance of instrumentation,

PROCEDURES AND PROGRAMS (Continued)

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. 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 NEI 94-01, Revision 3-A “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 2012 and the limitations and conditions specified in NEI 94-01, Revision 2-A, dated October 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 41.9 psig.

The maximum allowable containment leakage rate $L_a$, at $P_a$, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. 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 $\leq 0.06\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $< 0.75\ L_a$ for Type A tests;

2) Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.

g. Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments: Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during 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.

b. Provisions for performance criteria for SG tube integrity: SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or a combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG

PROCEDURES AND PROGRAMS (Continued)

Leakage is not to exceed 500 gpd per SG.

3. The operational LEAKAGE performance criterion is specified in RCS LCO 3.4.6.2, "Operational LEAKAGE."

c. Provisions for SG tube plugging criteria: Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth-based criteria:

1. Tubes with service-induced flaws located greater than 15.2 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.2 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections: Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. Portions of the tube below 15.2 inches below the top of the tubesheet are excluded from this requirement. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections)*. In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the

* As approved by License Amendment No. 277, inspection of the Millstone Unit 3 SGs A and C may be deferred, on a one-time basis, from fall 2020 (Refueling Outage 20) to spring 2022 (Refueling Outage 21).

PROCEDURES AND PROGRAMS (Continued)

inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;

b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

3. If crack indications are found in portions of the SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). 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

PROCEDURES AND PROGRAMS (Continued)

associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

h. 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 Ventilation System (CREVs), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

1. Appropriate application of ASTM E741 shall include the ability to take minor exceptions to the test methodology. These exceptions shall be documented in the test report, and

2. Vulnerability assessments for radiological, hazardous chemical and smoke, and emergency ventilation system testing were completed as documented in the UFSAR and other licensing basis documents. The exceptions to the Regulatory Guides (RG) referenced in RG 1.196 (i.e., RG 1.52, RG 1.78, and

PROCEDURES AND PROGRAMS (Continued)

RG 1.183), which were considered in completing the vulnerability assessments, are documented in the UFSAR/current licensing basis. Compliance with these RGs is consistent with the current licensing basis as described in the UFSAR and other licensing basis documents.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVs, operating at the flow rate required by the Surveillance Requirements, at a Frequency of 48 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

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 c. and d., respectively.

i. Snubber Examination, Testing, and Service Life Monitoring Program Plan

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

a. This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.

b. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b), subject to its limitations and modifications, and subject to Commission approval.

PROCEDURES AND PROGRAMS (Continued)

c. The program shall, as allowed by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements" and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants" in lieu of Section XI of the ASME BPV Code ISI requirements for snubbers, or meet authorized alternatives pursuant to 10 CFR 50.55a(a)(3).

d. The 120-month program updates shall be made in accordance with 10 CFR 50.55a (including 10 CFR 50.55a(b)(3)(v)) subject to the limitations and modifications listed therein.

j. Surveillance Frequency Control Program

This program provides controls for surveillance frequencies. The program shall ensure that surveillance requirements specified in the technical specification are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of frequencies of those surveillance requirements for which the frequency is controlled by the program.

b. Changes to the frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the frequencies established in the Surveillance Frequency Control Program.

6.8.5 Written procedures shall be established, implemented and maintained covering Section I.E, Radiological Environmental Monitoring, of the REMODCM.

6.8.6 All procedures and procedure changes required for the Radiological Environmental Monitoring Program (REMP) of Specification 6.8.5 above shall be reviewed by an individual (other than the author) from the organization responsible for the REMP and approved by appropriate supervision.

Temporary changes may be made provided the intent of the original procedure is not altered and the change is documented and reviewed by an individual (other than the author) from the organization responsible for the REMP, within 14 days of implementation.

## 6.9 REPORTING REQUIREMENTS

### ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector, unless otherwise noted.

### STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

### ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted in accordance with 10 CFR 50.4.

6.9.1.2a. Deleted

6.9.1.2b. The results of specific activity analyses in which the reactor coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioiodine isotope concentration (μCi/gm) as a function of time for the

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

ANNUAL REPORTS (Continued)

duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit. The report covering the previous calendar year shall be submitted prior to March 1 of each year.

6.9.1.3 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

NOTE

A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM), and in 10 CFR Part 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the REMODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in the next annual report.

6.9.1.4 RADIOACTIVE EFFLUENT RELEASE REPORT

NOTE

A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the REMODCM and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

MONTHLY OPERATING REPORTS

6.9.1.5 Deleted

CORE OPERATING LIMITS REPORT

6.9.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Reactor Core Safety Limit for Specification 2.1.1.
2. Overtemperature $\Delta T$ and Overpower $\Delta T$ setpoint parameters for Specification 2.2.1.
3. SHUTDOWN MARGIN for Specifications 3/4.1.1.1.1, 3/4.1.1.1.2, and 3/4.1.1.2.
4. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.
5. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.
6. Control Bank Insertion Limits for Specification 3/4.1.3.6.
7. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1.1.
8. Heat Flux Hot Channel Factor Limits for Specification 3/4.2.2.1.
9. RCS Total Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3.1.
10. DNB Parameters for Specification 3/4.2.5.
11. Shutdown Margin Monitor minimum count rate for Specification 3/4.3.5.
12. Boron Concentration for Specification 3/4.9.1.1.

CORE OPERATING LIMITS REPORT (Cont.)

6.9.1.6.b The analytical methods used to determine the core operating limits in Specification 6.9.1.6.a shall be those previously reviewed and approved by the NRC and identified below. The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number, title, revision, date, and any supplements).

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary). Methodology for Specifications:

    - 2.1.1 Reactor Core Safety Limits
    - 3.1.1.1.1 SHUTDOWN MARGIN – MODE 1 and 2
    - 3.1.1.1.2 SHUTDOWN MARGIN – MODES 3, 4 and 5 Loops Filled
    - 3.1.1.2 SHUTDOWN MARGIN – Cold Shutdown – Loops Not Filled
    - 3.1.1.3 Moderator Temperature Coefficient
    - 3.1.3.5 Shutdown Bank Insertion Limits
    - 3.1.3.6 Control Bank Insertion Limits
    - 3.2.1.1 AXIAL FLUX DIFFERENCE
    - 3.2.2.1 Heat Flux Hot Channel Factor
    - 3.2.3.1 RCS Total Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.9.1.1 REFUELING Boron Concentration
    - 3.2.5 DNB Parameters
    - 3.3.5 Shutdown Margin Monitor

2. Deleted

3. Deleted

4. WCAP-10216-P-A-R1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," (W Proprietary). (Methodology for Specifications 3.2.1.1--AXIAL FLUX DIFFERENCE and 3.2.2.1--Heat Flux Hot Channel Factor)

5. WCAP-16996-P-A, "REALISTIC LOCA EVALUATION METHODOLOGY APPLIED TO THE FULL SPECTRUM OF BREAK SIZES (FULL SPECTRUM LOCA METHODOLOGY)," (W Proprietary) (Methodology for Specification 3.2.2.1--Heat Flux Hot Channel Factor.)

CORE OPERATING LIMITS REPORT (Cont.)

6. WCAP-16009-P-A, "REALISTIC LARGE-BREAK LOCA EVALUATION METHODOLOGY USING THE AUTOMATED STATISTICAL TREATMENT OF UNCERTAINTY METHOD (ASTRUM)," (W Proprietary). (Methodology for Specification 3.2.2.1--Heat Flux Hot Channel Factor.)

7. WCAP-11946, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," (W Proprietary). Methodology for Specification:

    - 3.1.1.3 - Moderator Temperature Coefficient

8. WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

9. WCAP-10079-P-A, "NOTRUMP - A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

10. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

11. Deleted

12. Deleted

13. Deleted

14. Deleted

15. Deleted

16. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis." Methodology for Specification:

    - 3.2.2.1 - Heat Flux Hot Channel Factor

17. WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model." Methodology for Specification:

    - 3.2.2.1 - Heat Flux Hot Channel Factor

CORE OPERATING LIMITS REPORT (Cont.)

18. WCAP-8745-P-A, "Design Bases for the Thermal Overpower $\Delta T$ and Thermal Overtemperature DT Trip Functions," (Westinghouse Proprietary Class 2). (Methodology for Specifications 2.2.1 -- Overtemperature $\Delta T$ and Overpower $\Delta T$ Setpoints.)

19. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," (W Proprietary). (Methodology for Specification 3.2.2.1 - Heat Flux Hot Channel Factor.)

20. VEP-FRD-42-A, "Reload Nuclear Design Methodology." Methodology for Specifications:

    - 2.1.1 Reactor Core Safety Limits
    - 3.1.1.1.1 SHUTDOWN MARGIN – MODE 1 and 2
    - 3.1.1.1.2 SHUTDOWN MARGIN – MODES 3, 4 and 5 Loops Filled
    - 3.1.1.2 SHUTDOWN MARGIN – Cold Shutdown – Loops Not Filled
    - 3.1.1.3 Moderator Temperature Coefficient
    - 3.1.3.5 Shutdown Bank Insertion Limits
    - 3.1.3.6 Control Bank Insertion Limits
    - 3.2.2.1 Heat Flux Hot Channel Factor
    - 3.2.3.1 Nuclear Enthalpy Rise Hot Channel Factor
    - 3.3.5 Shutdown Margin Monitor
    - 3.9.1.1 REFUELING Boron Concentration

21. VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications." Methodology for Specifications:

    - 3.2.1.1 AXIAL FLUX DIFFERENCE
    - 3.2.2.1 Heat Flux Hot Channel Factor

22. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology." Methodology for Specifications:

    - 3.2.3.1 RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
    - 3.2.5 DNB Parameters

CORE OPERATING LIMITS REPORT (Cont.)

23. DOM-NAF-2-P-A, “Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code,” including Appendix C, “Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code,” and Appendix D, “Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code.” Methodology for Specifications:

- 3.2.3.1 RCS Flow Rate, Nuclear Enthalpy Rise Hot Channel Factor
- 3.2.5 DNB Parameters

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.7 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.8.4.g, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date and the effective plugging percentage in each steam generator.

STEAM GENERATOR TUBE INSPECTION REPORT (Continued)

g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

h. The primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

i. The calculated accident induced leakage rate from the portion of the tubes below 15.2 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.49 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined; and

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator Region I, and one copy to the NRC Resident Inspector, within the time period specified for each report.

6.10 Deleted.

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 Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of 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 possess:

   1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

   2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device’s dose alarm setpoint is reached, with an appropriate alarm setpoint, or

   3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

6.12 HIGH RADIATION AREA (cont.)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.12.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designees, and

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of

6.12 HIGH RADIATION AREA (cont.)

radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

## 6.13 RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

a. The REMODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental program; and

b. The REMODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release, reports required by Specification 6.9.1.3 and Specification 6.9.1.4.

Licensee initiated changes to the REMODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    1) sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

    2) a determination that the change(s) 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 50, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance by FSRC and the approval of the designated officer; and

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire REMODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the REMODCM 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 indicated the date (i.e., month and year) the change was implemented.

## 6.14 RADIOACTIVE WASTE TREATMENT

Procedures for liquid and gaseous radioactive effluent discharges from the Unit shall be prepared, approved, maintained and adhered to for all operations involving offsite releases of radioactive effluents. These procedures shall specify the use of appropriate waste treatment systems utilizing the guidance provided in the REMODCM.

The Solid Radioactive Waste Treatment System shall be operated in accordance with the Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

## 6.15 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the REMODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the REMODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the REMODCM;

d. 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 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the REMODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the REMODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be in accordance with the following:

    1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

    2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the SITE BOUNDARY, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of Specification 4.0.2 and Specification 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

## 6.16 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 REMODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the REMODCM.

b. 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

c. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

## 6.17 REACTOR COOLANT PUMP FLYWHEEL INSPECTION PROGRAM

This program shall provide for the inspection of each reactor coolant pump flywheel by either qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years.

## 6.18 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications:

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

    1. A change in the TS incorporated in the license or

    2. A change to updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.18.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

## 6.19 COMPONENT CYCLIC OR TRANSIENT LIMIT

This program provided controls to track the FSAR, Section 3.9N, cyclic and transient occurrences to ensure that components are maintained within the design limits.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. NPF-49

MILLSTONE POWER STATION, UNIT 3

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-423

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

MILLSTONE POWER STATION, UNIT NO. 3

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

TABLE OF CONTENTS

## 1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of non-radiological 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 of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters are regulated by way of the licensee's NPDES permit.

## 2.0 Environmental Protection Issues

In the FES-OL dated December, 1984, the staff considered the environmental impacts associated with the operation of Millstone Power Station, Unit No. 3. No environmental issues were identified which required study or license conditions to resolve environmental concerns and to assure adequate protection of the environment.

## 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 onsite 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

---

* This provision does not relieve the licensee of the requirements of 10 CFR 50.50.

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.

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 Permits or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The licensee shall notify the NRC of changes to the effective 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 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 but not included in the reporting requirements of 10 CFR 50.72, shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions, and unanticipated or emergency discharge of non-radioactive waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

## 4.2 Environmental Monitoring

### 4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decisions made by the State of Connecticut, under the authority of the Clean Water Act, for any requirements for aquatic monitoring.

### 4.2.2 Terrestrial Monitoring

No terrestrial monitoring is required.

## 5.0 Administrative Procedures

### 5.1 Review

The licensee shall provide for review of compliance with the EPP. The reviews shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review function and results of the review activities shall be maintained and made available for inspection.

### 5.2 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the station. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

## 5.3 Changes in Environmental Protection Plan

Requests for changes in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

## 5.4 Plant Reporting Requirements

### 5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC on or before May 1 of each year. The initial report shall be submitted on or before May 1 of the year following start of commercial operation of the plant.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of this EPP for the report period, including a comparison with related preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends toward irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of mitigating action.

The Annual Environmental Operating Report shall also include:

(1) A list of EPP noncompliances and the corrective actions taken to remedy them.

(2) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental question.

(3) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing results shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact, and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this Subsection. The NRC shall be provided with a copy of such report at the same time it is submitted to the other agency.

## APPENDIX C

## ADDITIONAL CONDITIONS
## OPERATING LICENSE NO. NPF-49

The licensee shall comply with the following conditions on the schedules noted below:

| Amendment Number | Additional Condition | Condition Completion Date |
|---|---|---|
| 162 | Millstone Unit No. 3 will incorporate the changes into the Final Safety Analysis Report (FSAR) as requested by letter dated January 22, 1998, as supplemented by letter dated July 17, 1998, that accepts the use of epoxy coatings on service water system components. Future changes to the design described in this submittal may be made in accordance with the provisions of 10 CFR 50.59. | During the next revision of the FSAR required by 10 CFR 50.71(e) or no later than June 30, 1999. |
| 164 | Millstone Unit No. 3 will incorporate into Technical Specification 6.9.1.6, references to the shutdown margin analysis methods reviewed and approved by the NRC. | To be submitted to the NRC within 90 days from October 21, 1998. |