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

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

(MILLSTONE POWER STATION, UNIT NO. 2)

RENEWED FACILITY OPERATING LICENSE

Renewed License No. DPR-65

1. The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-65 issued on September 26, 1975 has now found that:

   A. The application to renew License DPR-65 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 Part 54 Chapter 1, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Millstone Power Station, Unit 2, (facility) has been substantially completed in conformity with Construction Permit No. CPPR-76 and the application, as amended, the provisions of the Act and the rules and 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 accord with the Act and the Commission's regulations;

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

   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 rules and regulations of the Commission;

F. The licensee is technically and financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

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

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

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Renewed Facility Operating License No. DPR-65, subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 (formerly Appendix D to Part 50), 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, and 70, including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31.

2. On the basis of the foregoing findings regarding this facility, Facility Operating License No. DPR-65, issued on September 26, 1975, is superceded by Renewed Facility Operating License No. DPR-65, 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. 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by DENC. The facility is located on the north shore of Long Island Sound and on the east side of Niantic Bay in the Town of Waterford, Connecticut, about three miles from New London, Connecticut, and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 13 through 42), and the Environmental Report as amended (Amendments 1 through 5).

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Dominion Energy Nuclear Connecticut, Inc.:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location on the north shore of Long Island Sound and on the east side of Niantic Bay, in the Town of Waterford, Connecticut, about three miles from New London,

Connecticut, in accordance with the procedures and limitations set forth in this renewed operating license;

(2) 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) 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) 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 and equipment calibration or associated with radioactive apparatus or components;

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and 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

The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 2700 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 346 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report and as approved in the SER dated September 19, 1978, and supplements dated October 21, 1980, November 11, 1981, October 31, 1985, April 15, 1986, January 15, 1987, April 29, 1988, July 17, 1990, and November 3, 1995, 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.

(4) Physical Protection

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. 309, as supplemented by a change approved by License Amendment No. 323.

(5) Relocated Technical Specifications

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

a. This license condition approves the relocation of certain technical specification requirements to licensee-controlled documents (Technical Requirements Manual), as described in the licensee's application dated May 20, 1997, as supplemented on September 23, 1997. The approval is documented in the staff's safety evaluation dated November 19, 1997. This license condition is effective as of its date of issuance by Amendment No. 210 and shall be implemented 90 days from the date of issuance. Implementation shall include the relocation of technical specification requirements to the appropriate licensee-controlled document as identified in the licensee's application dated May 20, 1997, as supplemented on September 23, 1997.

(6) 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 transferof the licenses for MPS Unit No. 2.

(7) Immediately after the transfer of MPS Unit No. 2 to DNC*, the amount in the decommissioning trust for MPS Unit No. 2 must, with respect to the interest in MPS Unit No. 2 that DNC* would then hold, be at a level noless than the formula amount under 10 CFR50.75.

(8) The decommissioning trust agreement for MPS Unit No. 2 at the timethe transfer of the unit to DNC* is effected and thereafter is subject to the following:

   (a) The decommissioning trust agreement must be in aform 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. 2 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 Officeof 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 can not be amended in any material respectwithout 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

   (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'sregulations.

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

(9) 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. 2 license and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order.

(10) 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.

(11) 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 July 31, 2015, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

(12) 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.

(13) Mitigation Strategy License Condition

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

(14) Upon implementation of Amendment No. 305 adopting TSTF–448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 4.7.6.1.h, in accordance with TS 6.27.c.(i), the assessment of CRE habitability as required by TS 6.27.c.(ii), and the measurement of CRE pressure as required by TS 6.27.d, shall be considered met. Following implementation:

(a) The first performance of SR 4.7.6.1.h, in accordance with TS 6.27.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from November 2, 2006, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, TS 6.27.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from November 2, 2006, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic assessment of CRE habitability, TS 6.27.d, shall be within 3 years, plus the 9 month allowance of SR 4.0.2, from the date of the last tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(15) <u>Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"</u>

(a) The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; and the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA Sa 2009; as specified in Unit 2 License Amendment No. 337 dated January 30, 2020.

(b) The licensee will review the completed 10 CFR 50.54(f) reevaluation of external floods and update its 10 CFR 50.69 categorization procedures, as necessary, prior to the adoption of 10 CFR 50.69 to ensure that the potential for external flooding will be incorporated into the categorization process consistent with applicable guidance.

(c) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

D. This renewed operating license is effective as of its date of issuance and shall expire at midnight July 31, 2035.

FOR THE NUCLEAR REGULATORY COMMISSION

/ RA /

J. E. Dyer, Director

Office of Nuclear Reactor Regulation

Attachment:

1. Appendix A - Technical Specifications

Date of Issuance: November 28, 2005

MILLSTONE NUCLEAR POWER STATION
UNIT 2

TECHNICAL SPECIFICATIONS

AUG 1 1975

APPENDIX "A"
TO
LICENSE NO. DPR - 65

ISSUED BY THE UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSE AUTHORITY FILE COPY

DO NOT REMOVE

MILLSTONE NUCLEAR POWER STATION

UNIT 2

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

LICENSE NO. DPR-65

PAGES I THROUGH XVIII HAVE BEEN INTENTIONALLY DELETED

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

DEFINED TERMS

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

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

OPERATIONAL MODE

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

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principal specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 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, normal and emergency electrical power sources, 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).

REPORTABLE EVENT

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

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

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

1.8.2 The equipment hatch is closed and sealed, and

1.8.3 The airlock is in compliance with the requirements of Specification 3.6.1.3.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

---

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

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

SHUTDOWN MARGIN

1.13 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 control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

LEAKAGE

1.14 LEAKAGE shall be:

1.14.1 CONTROLLED LEAKAGE

CONTROLLED LEAKAGE shall be the water flow from the reactor coolant pump seals, and

1.14.2 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);

1.14.3 PRESSURE BOUNDARY LEAKAGE

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

1.14.4 UNIDENTIFIED LEAKAGE

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

AZIMUTHAL POWER TILT - T_q

1.18 AZIMUTHAL POWER TILT shall be the difference between the maximum power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

$$\text{AZIMUTHALPOWERTILT} = \left[\frac{\text{Maximum power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}}\right] - 1$$

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro-curie/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, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

DOSE EQUIVALENT XE-133

1.20 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (micro-curie/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."

1.21 Deleted

FREQUENCY NOTATION

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

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) used for normal control and indication is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = AY_E + B$$

CORE OPERATING LIMITS REPORT

1.24 The CORE OPERATING LIMITS REPORT is the unit specific document that provides the 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 6.9.1.8. Plant operation within these operating limits is addressed in individual specifications.

ENCLOSURE BUILDING INTEGRITY - DELETED

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.27 The ENGINEERED SAFETY FEATURE 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

ENGINEERED SAFETY FEATURE RESPONSE TIME (Continued)

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.

PHYSICS TESTS

1.28 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r

1.29 The TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core. This value includes the effect of AZIMUTHAL POWER TILT.

SOURCE CHECK

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

PURGE - PURGING

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

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	> 5%	≥ 300°F
2. STARTUP	≥ 0.99	≤ 5%	≥ 300°F
3. HOT STANDBY	< 0.99	0	≥ 300°F
4. HOT SHUTDOWN	< 0.99	0	300°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.98	0	≤ 200°F
6. REFUELING**	≤ 0.95	0	≤ 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.

VENTING

1.35 VENTING is 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.

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its 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.

SITE BOUNDARY

1.37 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

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 industrial, commercial institutional and/or recreational purposes.

STORAGE PATTERN

1.39 A STORAGE PATTERN designates acceptable fuel assembly storage in a 2 x 2 storage array (4 spent fuel rack storage locations) within Regions 1, 2, and 4 of the spent fuel racks. Each 2 x 2 storage array includes at least one location in which storage is NOT permitted (fuel or non-fuel).

1.40 A NON-STANDARD FUEL CONFIGURATION is an object containing fuel that does not conform to the standard fuel configuration. The standard fuel configuration is a 14 x 14 array of fuel rods (or fuel rods replaced by un-enriched fuel rods or stainless steel rods) with five (5) guide tubes that occupy four lattice pitch locations each. Fuel in any other array is a "Non-standard Fuel Configuration." Reconstituted fuel in which one or more fuel rods have been replaced by either un-enriched fuel rods or stainless steel rods is considered to be a standard fuel configuration.

TABLE 1.2
FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Prior to each release.
N.A.	Not applicable.
SFCP	At the frequency specified in the Surveillance Frequency Control Program.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, 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.

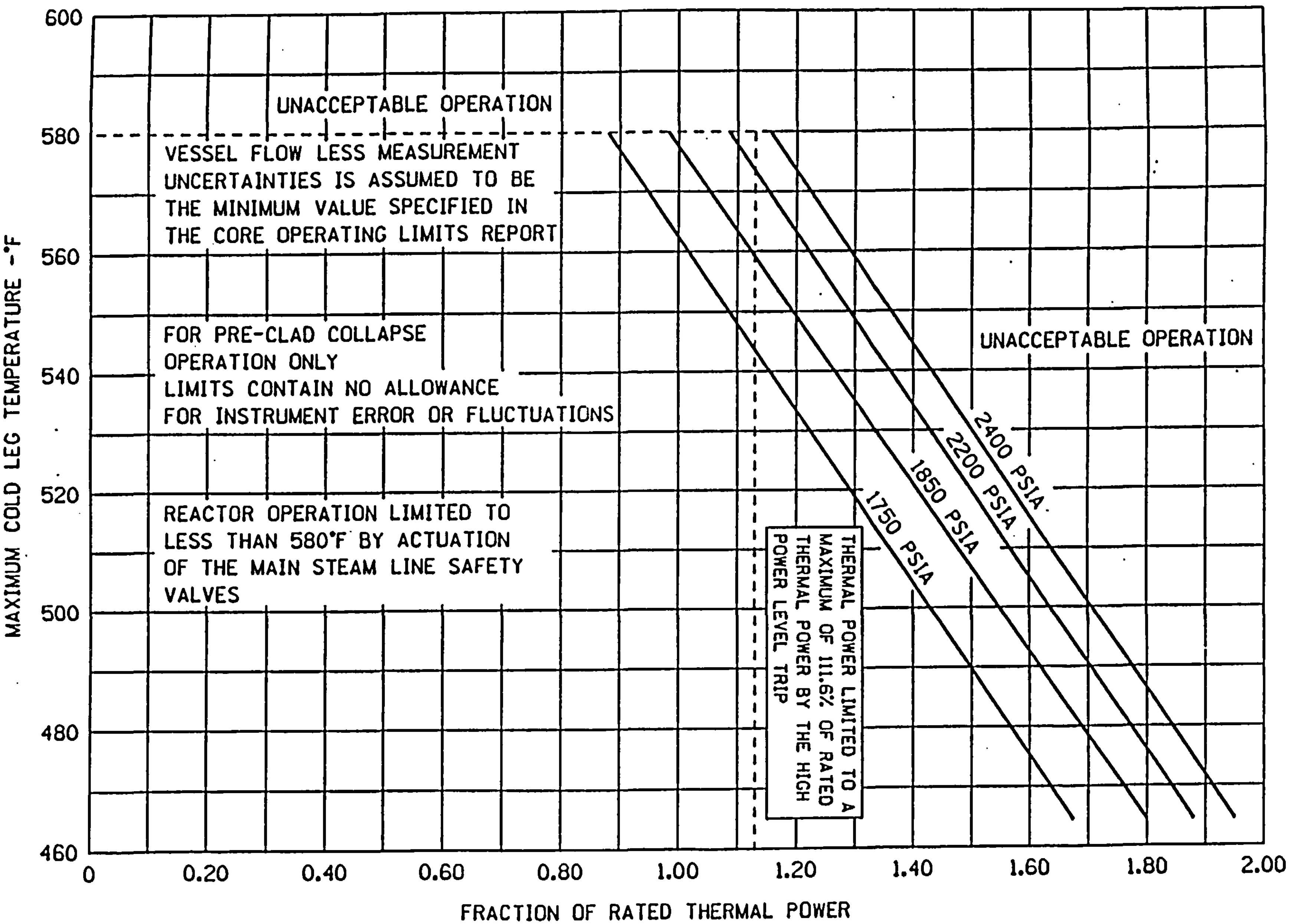


FIGURE 2.1-1
REACTOR CORE THERMAL MARGIN SAFETY LIMIT-
FOUR REACTOR COOL PUMPS OPERATING

SKM-MCR- dgn

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

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

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

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

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Level-High		
	Four Reactor Coolant Pumps Operating	≤ 9.6% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER, and a maximum of ≤ 106.6% of RATED THERMAL POWER.	≤ 9.7% Above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3.	Reactor Coolant Flow - Low (1)	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.9% of reactor coolant flow with 4 pumps operating.
4.	DELETED		
5.	Pressurizer Pressure - High	≤ 2397 psia	≤ 2407 psia
6.	Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
7.	Steam Generator Pressure - Low (2) (5)	≥ 691 psia	≥ 677 psia
8.	Steam Generator Water Level - Low (5)	≥ 48.5% Water Level - each steam generator	≥ 47.5% Water Level - each steam generator
9.	Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

*Design Reactor Coolant flow with 4 pumps operating is the lesser of either:
a. The reactor coolant flow rate measured per Specification 4.2.6.1, or
b. The minimum value specified in the CORE OPERATING LIMITS REPORT.

TABLE 2.2-1
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10.	Thermal Margin/Low Pressure (1)		
	Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).
11.	Loss of Turbine--Hydraulic Fluid (3) Pressure - Low	≥ 500 psig	≥ 500 psig
12.	Wide Range Logarithmic Neutron Flux Monitor - Shutdown	Not Applicable	Not Applicable
13.	Reactor Protection System Logic Matrices	Not Applicable	Not Applicable
14.	Reactor Protection System Logic Matrix Relays	Not Applicable	Not Applicable
15.	Reactor Trip Breakers	Not Applicable	Not Applicable

TABLE NOTATION

(1) Trip may be bypassed when logarithmic power is < 1E-04% and the bypass shall be capable of automatic removal whenever logarithmic power is < 1E-04%. Bypass shall be removed prior to raising logarithmic power to a value ≥ 1E-04%.

(2) Trip may be manually bypassed when steam generator pressure is < 800 psia and all CEAs are fully inserted; bypass shall be automatically removed when steam generator pressure is ≥ 800 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(4) Calculations of the trip setpoint includes measurements, calculational and processor uncertainties, and dynamic allowances.

(5) Each of four channels actuate on the auctioneered output of two transmitters, one from each steam generator.

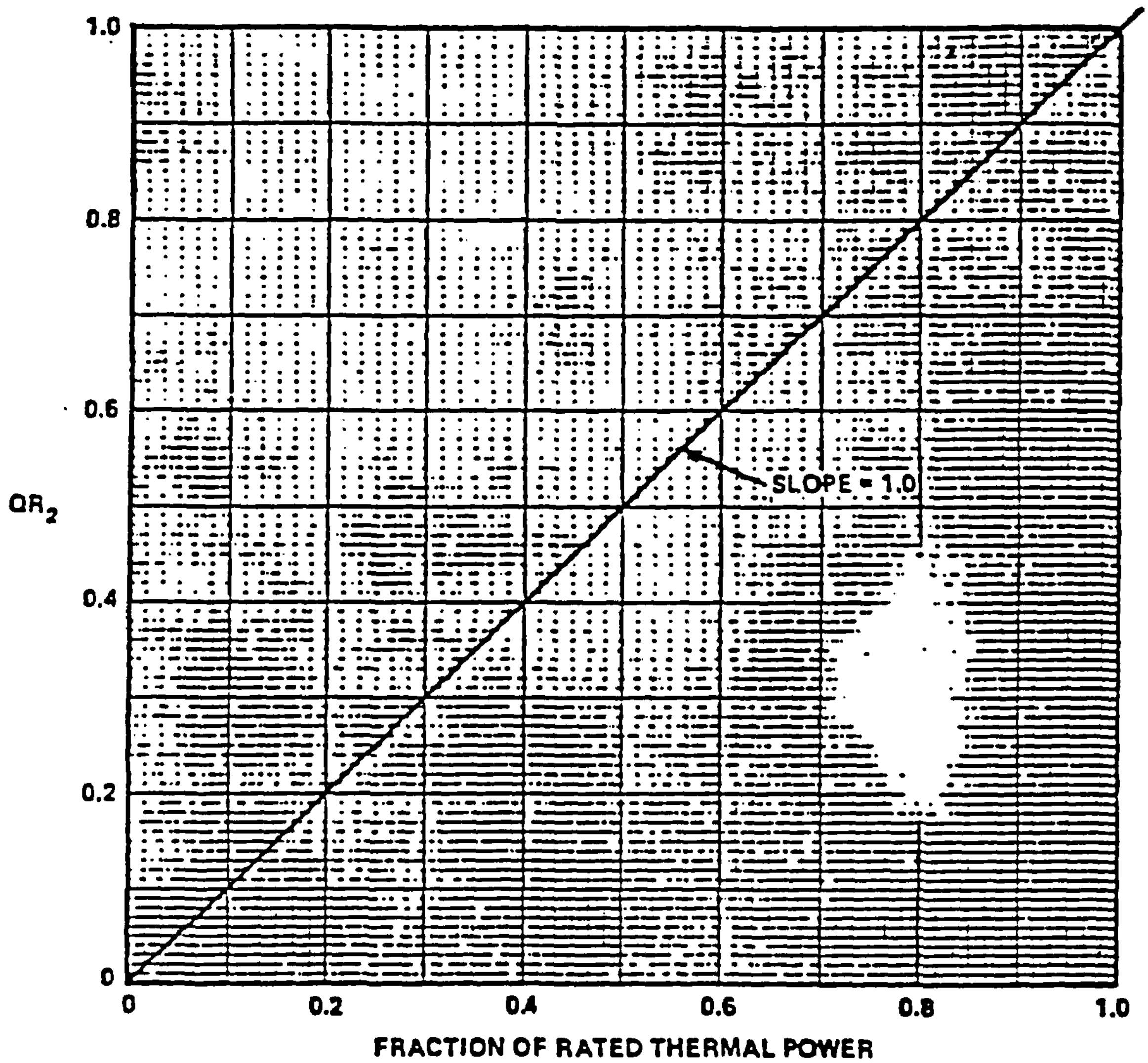


FIGURE 2.2-1

Local Power Density — High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

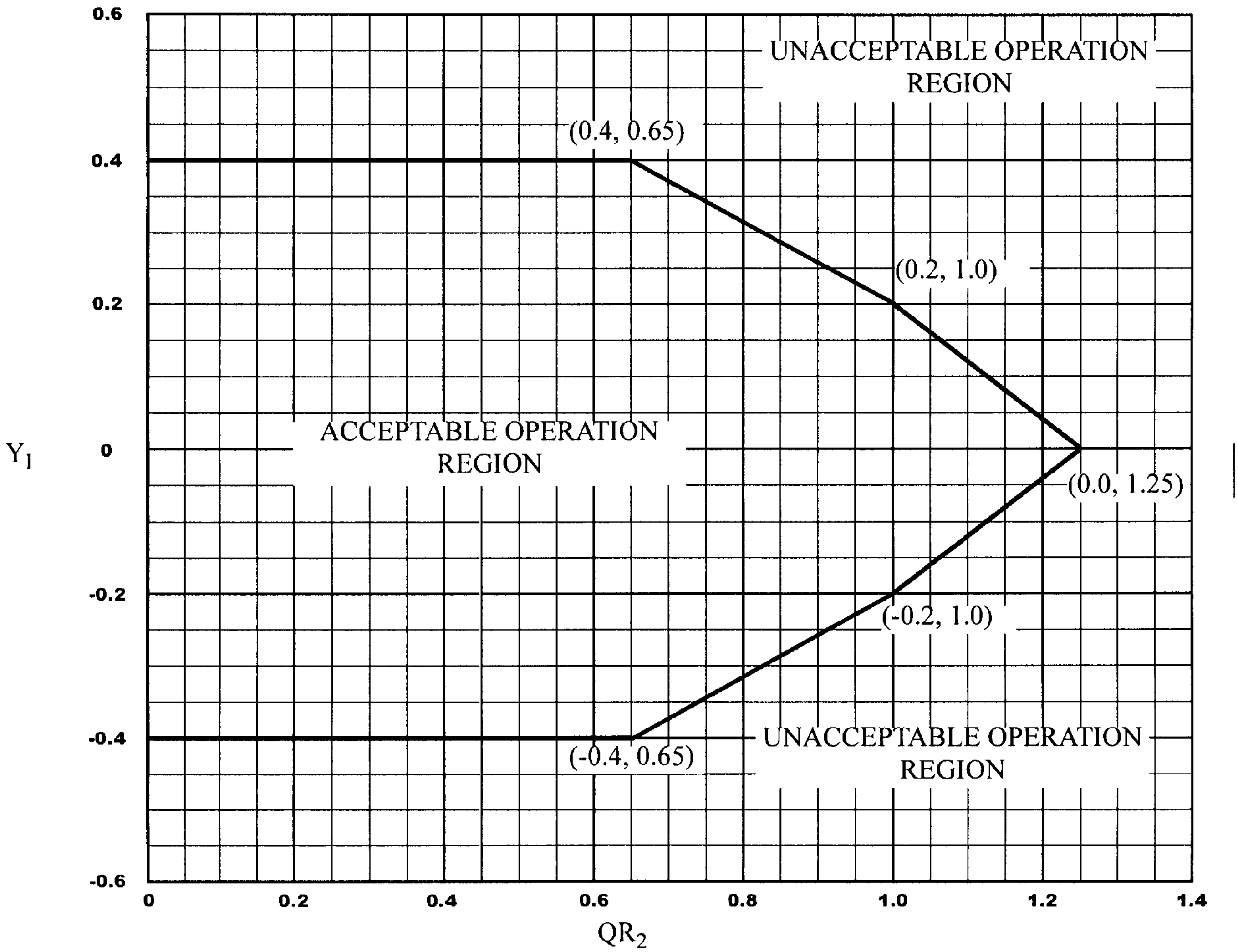


FIGURE 2.2-2 Local Power Density - High Trip Setpoint Part 2 (QR_2 Versus Y_I)

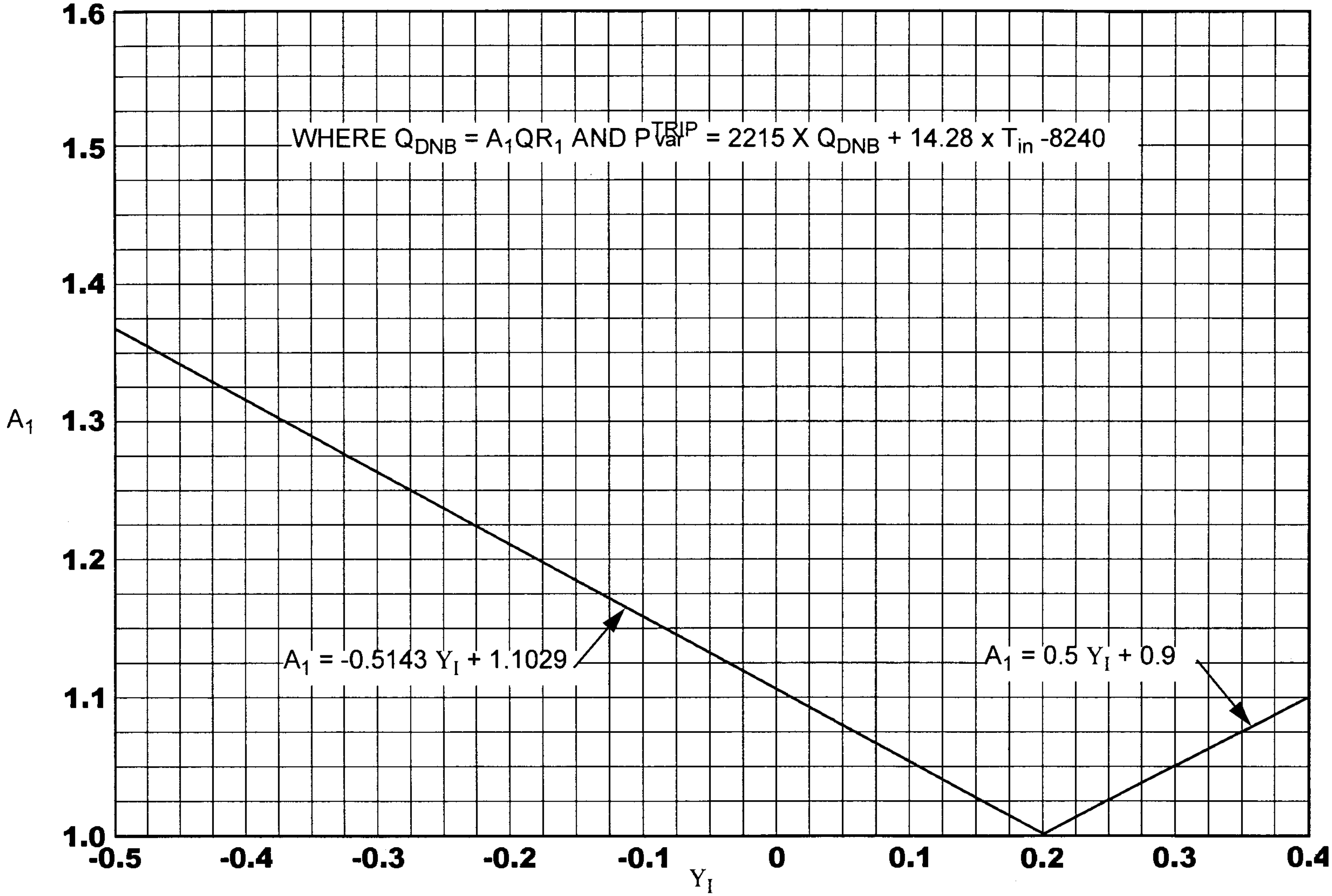


FIGURE 2.2-3 Thermal Margin/Low Pressure Trip Setpoint Part 1 (Y_I versus A_1)

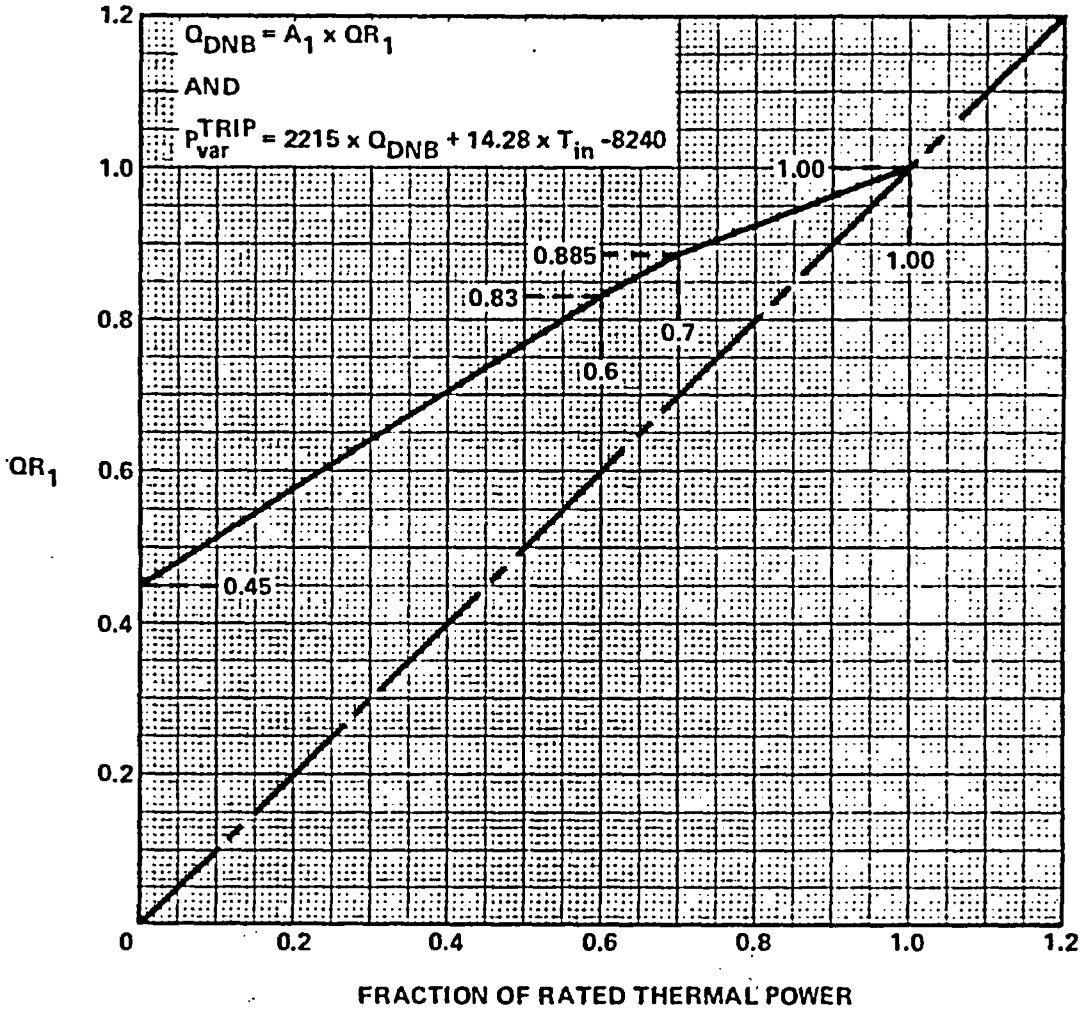


FIGURE 2.2-4 Thermal Margin/Low Pressure Trip Setpoint (Part 2 Fraction of RATED THERMAL POWER Versus QR_1)

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.

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 LCO 3.0.6. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

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

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

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time it is identified that a Limiting Condition for Operation is not met. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODES 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 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within 2 hours, ACTION shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:

APPLICABILITY:

LIMITING CONDITION FOR OPERATION (Continued)

1. At least HOT STANDBY within the next 6 hours.

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

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

This specification is not applicable in MODES 5 or 6.

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

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

Failure to meet a Surveillance, 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.

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 time 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 of 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.

APPLICABILITY:

SURVEILLANCE REQUIREMENTS (Continued)

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
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.

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 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - (SDM)

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 3$^{(1)*}$, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN not within the limit specified in the CORE OPERATING LIMITS REPORT, within 15 minutes, initiate and continue boration at ≥ 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the SHUTDOWN MARGIN is restored to within limit.

SURVEILLANCE REQUIREMENTS

4.1.1.1 Verify SHUTDOWN MARGIN is within the limit specified in the CORE OPERATING LIMITS REPORT at the frequency specified in the Surveillance Frequency Control Program.

---

*$^{(1)}$See Special Test Exception 3.10.1

"This page left intentionally blank."

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 REACTIVITY CONTROL SYSTEMS

REACTIVITY BALANCE

LIMITING CONDITION FOR OPERATION

3.1.1.2 The core reactivity balance shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTION:

With core reactivity balance not within limit:

Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation and establish appropriate operating restrictions and Surveillance Requirements within 7 days or otherwise be in MODE 3 within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.2 Verify*(1) overall core reactivity balance is within ± 1% Δk/k of predicted values prior to entering MODE 1 after fuel loading and at the frequency specified in the Surveillance Frequency Control Program**(2). The provisions of Specification 4.0.4 are not applicable.

*(1) The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

**(2) Only required after 60 Effective Full Power Days.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The following boron dilution restrictions shall be met:

a. The flow rate of reactor coolant through the core shall be ≥ 1000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

b. A maximum of two charging pumps shall be capable of injecting into the Reactor Coolant System whenever the temperature of one or more of the Reactor Coolant System cold legs is < 300°F.

APPLICABILITY: ALL MODES.

ACTION:

a. With the flow rate of reactor coolant through the core < 1000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

b. With more than two charging pumps capable of injecting into the Reactor Coolant System and the temperature of one or more of the Reactor Coolant System cold legs is < 300°F, take immediate action to comply with 3.1.1.3.b.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1* The reactor coolant flow rate through the core shall be determined to be ≥ 1000 gpm prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 1000 gpm through the core.

4.1.1.3.2 One charging pump shall be demonstrated not capable of injecting into the Reactor Coolant System at least once per 12 hours whenever the temperature of one or more of the Reactor Coolant System cold legs is < 300°F.

---

* When the plant is in MODE 1 or 2, reactor coolant pumps are required to be in operation. Therefore, Surveillance Requirement 4.1.1.3.1 does not have to be performed in MODES 1 and 2.

MODERATOR TEMPERATURE COEFFICIENT (MTC)

LIMITING CONDITION FOR OPERATION (Continued)

3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT. The upper limit shall be less than or equal to:

a. 0.7×10^{-4} $\Delta K/K/^{\circ}F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. 0.4×10^{-4} $\Delta K/K/^{\circ}F$ whenever THERMAL POWER IS > 70% of RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENT

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparision with the predicted values.

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

#See Special Test Exemption 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each refueling.

b. At any THERMAL POWER, within 14 EFPD after each fuel loading at equilibrium boron concentration.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be ≥ 515°F when the reactor is critical.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the Reactor Coolant System temperature (T_{avg}) < 515°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be ≥ 515°F.

a. Within 15 minutes prior to making the reactor critical, and

b. At the frequency specified in the Surveillance Frequency Control Program when the reactor is critical and the Reactor Coolant System temperature (T_{avg}) is < 525°F.

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

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 INTENTIONALLY LEFT BLANK

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All CEAs shall be OPERABLE with each CEA of a given group positioned within 10 steps (indicated position) of all other CEAs in its group, and the CEA Motion Inhibit and the CEA Deviation Circuit shall be OPERABLE.

APPLICABILITY: MODES 1[(1)] and 2[(1)].

ACTION:

INOPERABLE EQUIPMENT	REQUIRED ACTION
A. One or more CEAs trippable and misaligned from all other CEAs in its group by > 10 steps and < 20 steps. OR One CEA trippable and misaligned from all other CEAs in its group by ≥ 20 steps.	A.1 Reduce THERMAL POWER to < 70% of the maximum allowable THERMAL POWER within 1 hour and restore CEA(s) misalignment within 2 hours or otherwise be in MODE 3 within the next 6 hours.
B. CEA Motion Inhibit inoperable.	B.1 Verify the indicated position of each CEA to be within 10 steps of all other CEAs in its group within 1 hour and every 4 hours thereafter, and restore CEA Motion Inhibit to OPERABLE status within 6 hours or otherwise be in MODE 3 within the next 6 hours. OR B.2[(2)] Place and maintain the CEA drive system mode switch in either the "off" or "manual" position, and withdraw all CEAs in group 7 to ≥ 172 steps within 6 hours or otherwise be in MODE 3 within the next 6 hours.

(1) See Special Test Exception 3.10.2

(2) Performance of ACTION B.2 is allowed only when not in conflict with either Required Action A.1 or C.1.

ACTION: (Continued):

C. CEA Deviation Circuit inoperable.	C.1 Verify the indicated position of each CEA to be within 10 steps of all other CEAs in its group within 1 hour and every 4 hours thereafter or otherwise be in MODE 3 within the next 6 hours.
D. One or more CEAs untrippable. OR Two or more CEAs misaligned by ≥ 20 steps.	D.1 Be in MODE 3 within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 Verify the indicated position of each CEA to be within 10 steps of all other CEAs in its group at the frequency specified in the Surveillance Frequency Control Program AND within 1 hour following any CEA movement larger than 10 steps.

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

SR 4.1.3.1.2 is not required to be performed for CEA 39 for the remainder of Cycle 24.

4.1.3.1.2 Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core 10 steps in either direction at the frequency specified in the Surveillance Frequency Control Program.

4.1.3.1.3 Verify the CEA Deviation Circuit is OPERABLE at the frequency specified in the Surveillance Frequency Control Program by a functional test of the CEA group Deviation Circuit which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 10 steps (indicated position).

4.1.3.1.4 Verify the CEA Motion Inhibit is OPERABLE by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents regulating CEAs from being inserted beyond the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT:

a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 31 days, and

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

"This page intentionally left blank."

This page intentionally left blank.

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All shutdown and regulating CEA reed switch position indicator channels and CEA pulse counting position indicator channels shall be OPERABLE and capable of determining the absolute CEA positions within ± 3 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. Deleted.

b. With a maximum of one reed switch position indicator channel per group or one (except as permitted by ACTION item d. below) pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, within 4 hours either:

   1. Restore the inoperable position indicator channel to OPERABLE status, or

   2. Be in HOT STANDBY, or

   3. Reduce THERMAL POWER to ≤ 70% of the maximum allowable THERMAL POWER level; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Operation at or below this reduced THERMAL POWER level may continue provided that within the next 4 hours either:

      a) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications; or

LIMITING CONDITION FOR OPERATION (Continued)

b) The CEA group(s) with the inoperable indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.

4. If the failure of the position indicator channel(s) is during STARTUP, the CEA group(s) with the inoperable position indicator channel must be moved to the "Full Out" position and verified to be fully withdrawn via a "Full Out" indicator within 4 hours.

c. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel at either its fully inserted position or fully withdrawn position, operation may continue provided:

1. The position of this CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable).

2. The fully inserted CEA group(s) containing the inoperable position channel is subsequently maintained fully inserted, and

3. Subsequent operation is within the limits of Specification 3.1.3.6.

d. With one or more pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each required position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 6 steps at the frequency specified in the Surveillance Frequency Control Program.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual CEA drop time, from a fully withdrawn position, shall be ≤ 2.75 seconds from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time shall be demonstrated through measurement with $T_{avg} \geq 515°F$, and all reactor coolant pumps operating prior to reactor criticality:

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

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

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

REACTIVITY CONTROL SYSTEMS

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to ≥ 176 steps.

APPLICABILITY: MODE 1*(1)
MODE 2(1),(2)** with any regulating CEA not fully inserted.

ACTION:

INOPERABLE EQUIPMENT	REQUIRED ACTION
A. One or more shutdown CEAs not within limit.	A.1 Restore shutdown CEA(s) to within limit within 2 hours or otherwise be in MODE 3 within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Verify each shutdown CEA is withdrawn ≥ 176 steps at the frequency specified in the Surveillance Frequency Control Program.

*(1) This LCO is not applicable while performing Specification 4.1.3.1.2.

**(2)See Special Test Exceptions 3.10.1 and 3.10.2.

REACTIVITY CONTROL SYSTEMS

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The power dependent insertion limit (PDIL) alarm circuit shall be OPERABLE, and the regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY[1]: MODES 1[2] and 2[2],[3].

ACTION:

INOPERABLE EQUIPMENT	REQUIRED ACTION
A. Regulating CEA groups inserted beyond the Transient Insertion Limits provided in the CORE OPERATING LIMITS REPORT.	A.1 Restore regulating CEA groups to within limits specified in the CORE OPERATING LIMITS REPORT within 2 hours or otherwise be in MODE 3 within the next 6 hours. OR A.2 Reduce THERMAL POWER to less than or equal to the fraction of RATED THERMAL POWER allowed by the CEA group position and insertion limits specified in the CORE OPERATING LIMITS REPORT within 2 hours or otherwise be in MODE 3 within the next 6 hours.

[1] This LCO is not applicable while performing Specification 4.1.3.1.2.
[2] See Special Test Exceptions 3.10.1 and 3.10.2.

[3] With $K_{eff} \geq 1.0$

REGULATING CEA INSERTION LIMITS (Continued)

B. Regulating CEA groups inserted between the Long Term Steady State Insertion limit and the Transient Insertion Limit specified in the CORE OPERATING LIMITS REPORT for intervals > 4 hours per 24 hour interval.	B.1 Verify Short Term Steady State Insertion Limits as specified in the CORE OPERATING LIMITS REPORT are not exceeded within 15 minutes or otherwise be in MODE 3 within the next 6 hours. OR B.2 Restrict increases in THERMAL POWER to < 5% RATED THERMAL POWER per hour within 15 minutes or otherwise be in MODE 3 within the next 6 hours.
C. Regulating CEA groups inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit specified in the CORE OPERATING LIMITS REPORT for intervals > 5 effective full power days (EFPD) per 30 EFPD or interval > 14 EFPD per 365 EFPD.	C.1 Restore regulating CEA groups to within the Long Term Steady State Insertion Limit specified in the CORE OPERATING LIMITS REPORT within 2 hours or otherwise be in MODE 3 within the next 6 hours.
D. PDIL alarm circuit inoperable.	D.1 Perform Specification 4.1.3.6.1 within 1 hour and once per 4 hours thereafter or otherwise be in MODE 3 within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 Verify each regulating CEA group position is within the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT at the frequency specified in the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable for entering into MODE 2 from MODE 3.

4.1.3.6.2 Verify the accumulated times during which the regulating CEA groups are inserted beyond the Steady State Insertion Limits but within the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT at the frequency specified in the Surveillance Frequency Control Program.

4.1.3.6.3 Verify PDIL alarm circuit is OPERABLE at the frequency specified in the Surveillance Frequency Control Program.

THIS PAGE LEFT BLANK INTENTIONALLY

CONTROL ROD DRIVE MECHANISMS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod drive mechanisms shall be de-energized.

APPLICABILITY: MODES 3*, 4, 5 and 6, whenever the RCS boron concentration is less than refueling concentration of Specification 3.9.1.

ACTION:

With any of the control rod drive mechanisms energized, restore the mechanisms to their de-energized state within 2 hours or immediately open the reactor trip circuit breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod drive mechanisms shall be verified to be de-energized at the frequency specified in the Surveillance Frequency Control Program.

* The control rod drive mechanisms may be energized for MODE 3 as long as 4 reactor coolant pumps are OPERATING, the reactor coolant system temperature is greater than 500° F, the pressurizer pressure is greater than 2000 psia and the requirements of Limiting Condition for Operation for Specification 3.3.1.1, "Reactor Protective Instrumentation," are met.

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION (Continued)

3.2.1 The linear heat rate, including heat generated in the fuel, clad and moderator, shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1.

ACTION:

During operation with the linear heat rate being monitored by the Incore Detector Monitoring System, comply with the following ACTION:

With the linear heat rate exceeding the limit as indicated by four or more coincident incore channels, within 15 minutes initiate corrective action to reduce the linear heat rate to less than or equal to the limit and either:

a. Restore the linear heat rate to less than or equal to the limit within one hour, or

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

During operation with the linear heat rate being monitored by the Excore Detector Monitoring System, comply with the following ACTIONS:

With the linear heat rate exceeding its limit, as indicated by the AXIAL SHAPE INDEX being outside of the power dependent limits on the Power Ratio Recorder, either:

a. Restore the AXIAL SHAPE INDEX to within the limits specified in the CORE OPERATING LIMITS REPORT within 1 hour from initially exceeding the linear heat rate limit, or

b. Be in at least HOT STANDBY within the next 4 hours.

SURVEILLANCE REQUIREMENT

4.2.1.1 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.2 Excore Detector Monitoring System*(1) - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at the frequency specified in the Surveillance Frequency Control Program that the CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6.

b. Verifying at the frequency specified in the Surveillance Frequency Control Program that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the allowable limits specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 Incore Detector Monitoring System**(2),***(3) - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at the frequency specified in the Surveillance Frequency Control Program.

b. Have their alarm setpoint adjusted to less than or equal to the limits specified in the CORE OPERATING LIMITS REPORT at the frequency specified in the Surveillance Frequency Control Program.

*(1) Only required to be met when the Excore Detector Monitoring System is being used to determine Linear Heat Rate.

**(2) Only required to be met when the Incore Detector Monitoring System is being used to determine Linear Heat Rate.

***(3) Not required to be performed below 20% RATED THERMAL POWER.

FIGURE 3.2-1 LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

TI iS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

Correction Letter of 5-4-89

POWER DISTRIBUTION LIMITS

TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F^T_r shall be within the 100% power limit specified in the CORE OPERATING LIMITS REPORT. The F^T_r value shall include the effect of AZIMUTHAL POWER TILT.

APPLICABILITY: MODE 1 with THERMAL POWER >20% RTP*.

ACTION:

With F^T_r exceeding the 100% power limit within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_r to within the power dependent limit specified in the CORE OPERATING LIMITS REPORT and withdraw the CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F^T_r shall be determined to be within the 100% power limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At the frequency specified in the Surveillance Frequency Control Program in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.020.

4.2.3.3 F^T_r shall be determined by using the incore detectors to obtain a power distribution map with all CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump Combination.

---

* See Special Test Exception 3.10.2.

AZIMUTHAL POWER TILT - T_Q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall be ≤ 0.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% of RATED THERMAL POWER(1)*.

ACTION:

a. With the indicated T_q > 0.02 but ≤ 0.10, either restore T_q to ≤ 0.02 within 2 hours or verify the TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR (F^T_r) is within the limit of Specification 3.2.3 within 2 hours and once per 8 hours thereafter. Or otherwise, reduce THERMAL POWER to ≤ 50% of RATED THERMAL POWER within the next 4 hours.

b. With the indicated T_q > 0.10, perform the following actions: (2)**

1. Verify the TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR (F^T_r) is within the limit of Specification 3.2.3 within 2 hours; and

2. Reduce THERMAL POWER to ≤ 50% of RATED THERMAL POWER within 2 hours; and

3. Restore T_q ≤ 0.02 prior to increasing THERMAL POWER. 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 measured T_q is verified ≤ 0.02 at least once per hour for 12 hours, or until verified at 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4.1 Verify T_q is within limit at the frequency specified in the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable for entering into MODE 1 with THERMAL POWER > 50% of RATED THERMAL POWER from MODE 1.

*(1)See Special Test Exception 3.10.2.

**(2)All subsequent Required ACTIONS must be completed if power reduction commences prior to restoring T_q ≤ 0.10.

"This page intentionally left blank."

This page intentionally left blank.

POWER DISTRIBUTION LIMITS

DNB MARGIN

LIMITING CONDITION FOR OPERATION

3.2.6 The DNB margin shall be preserved by maintaining the cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.6.1 The cold leg temperature, pressurizer pressure, and AXIAL SHAPE INDEX shall be determined to be within the limits specified in the CORE OPERATING LIMITS REPORT at the frequency specified in the Surveillance Frequency Control Program. The reactor coolant flow rate shall be determined to be within the limit specified in the CORE OPERATING LIMITS REPORT at the frequency specified in the Surveillance Frequency Control Program.

4.2.6.2 The provisions of Specification 4.0.4 are not applicable.

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each required reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in
Table 4.3-1.

4.3.1.1.2 The bypass function and automatic bypass removal function shall be demonstrated OPERABLE during a CHANNEL FUNCTIONAL TEST once within 92 days prior to each reactor startup. The total bypass function shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at the frequency specified in the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function.

TABLE 3.3-1

REACTOR PROTECTIVE 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 and *	1
2.	Power Level - High	4	2(f)	3	1, 2, 3(d)	2, 7, 8
3.	Reactor Coolant Flow - Low	4	2(a)	3	1, 2	2, 7, 8
4.	Pressurizer Pressure - High	4	2	3	1, 2	2
5.	Containment Pressure - High	4	2	3	1, 2	2
6.	Steam Generator Pressure - Low	4	2(b)	3	1, 2	2, 7, 8
7.	Steam Generator Water Level - Low	4	2	3	1, 2	2
8.	Local Power Density - High	4	2(c)	3	1	2, 7, 8
9.	Thermal Margin/Low Pressure	4	2(a)	3	1, 2	2, 7, 8
10.	Loss of Turbine - Hydraulic Fluid Pressure - Low	4	2(c)	3	1	2

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	4	0	2	3, 4, 5	4
12. DELETED					
13. Reactor Protection System Logic Matrices	6	1	6	1, 2 and *	5
14. Reactor Protection System Logic Matrix Relays	4/Matrix	3/Matrix	4/Matrix	1, 2 and *	6
15. Reactor Trip Breakers	4	3	4	1, 2 and *	6

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(a) Trip may be bypassed when logarithmic power is < 1E-04% and the bypass shall be capable of automatic removal whenever logarithmic power is < 1E-04%. Bypass shall be removed prior to raising logarithmic power to a value ≥ 1E-04%.

(b) Trip may be manually bypassed when steam generator pressure is < 800 psia and all CEAs are fully inserted; bypass shall be automatically removed when steam generator pressure is ≥ 800 psia.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(d) Trip does not need to be OPERABLE if all the control rod drive mechanisms are de-energized or if the RCS boron concentration is greater than or equal to the refueling concentration of Specification 3.9.1.

(e) DELETED

(f) ΔT Power input to trip may be bypassed when logarithmic power is < 1E-04% and the bypass shall be capable of automatic removal whenever logarithmic power is < 1E-04%. Bypass shall be removed prior to raising logarithmic power to a value ≥ 1E-04%.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 4 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. The inoperable channel shall either be restored to OPERABLE status, or placed in the tripped condition, within 48 hours.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also declared inoperable, and the appropriate actions are taken for the affected functional units.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

ACTION 3 - NOT USED

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, immediately verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1, and at least once per 4 hours thereafter.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 6 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 7 – With one automatic bypass removal channel inoperable for one or more functions, either

a. disable the bypass channel within 1 hour, <u>or</u>

b. place the affected trip units in bypass or trip within 1 hour, <u>and</u> either

1. restore the bypass removal channel and affected trip units to OPERABLE status within 48 hours, <u>or</u>

2. place the affected trip units in trip within 48 hours.

ACTION 8 – With two automatic bypass removal channels inoperable for one or more functions, either

a. disable the bypass channels within 1 hour, <u>or</u>

b. place one affected trip unit in bypass and place the other in trip for each affected trip function, within 1 hour, <u>and</u>

restore one automatic bypass removal channel and the associated trip unit to OPERABLE status for each affected trip function, within 48 hours.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2.	Power Level - High				
	a. Nuclear Power	SFCP	SFCP(2), SFCP(3),SFCP(5)	SFCP	1, 2, 3*
	b. ΔT Power	SFCP	SFCP(4), SFCP	SFCP	1
3.	Reactor Coolant Flow - Low	SFCP	SFCP	SFCP	1, 2
4.	Pressurizer Pressure - High	SFCP	SFCP	SFCP	1, 2
5.	Containment Pressure - High	SFCP	SFCP	SFCP	1, 2
6.	Steam Generator Pressure - Low	SFCP	SFCP	SFCP	1, 2
7.	Steam Generator Water Level - Low	SFCP	SFCP	SFCP	1, 2
8.	Local Power Density - High	SFCP	SFCP	SFCP	1
9.	Thermal Margin/Low Pressure	SFCP	SFCP	SFCP	1, 2
10.	Loss of Turbine--Hydraulic Fluid Pressure - Low	N.A.	SFCP	S/U(1)	N.A.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
11.	Wide Range Logarithmic Neutron Flux Monitor - Shutdown	SFCP	SFCP(5)	S/U(1)	3, 4, 5
12.	DELETED				
13.	Reactor Protection System Logic Matrices	N.A.	N.A.	SFCP and S/U(1)	1, 2 and *
14.	Reactor Protection System Logic Matrix Relays	N.A.	N.A.	SFCP and S/U(1)	1, 2 and *
15.	Reactor Trip Breakers	N.A.	N.A.	SFCP	1, 2 and *

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometers to make nuclear power signals agree with calorimetric calculation. During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to $\leq$ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Above 15% of RATED THERMAL POWER, adjust "ΔT Pwr Calibrate" potentiometers to null "Nuclear Pwr - ΔT Pwr". During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(5) Neutron detectors are excluded from the CHANNEL CALIBRATION.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The engineered safety feature actuation system instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an engineered safety feature actuation system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either adjust the trip setpoint to be consistent with the value specified in the Trip Setpoint column of Table 3.3-4 within 2 hours or declare the channel inoperable and take the ACTION shown in Table 3.3-3.

b. With an engineered safety feature actuation system instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each required engineered safety feature actuation system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The bypass function and automatic bypass removal function shall be demonstrated OPERABLE during a CHANNEL FUNCTIONAL TEST once within 92 days prior to each reactor startup. The total bypass function shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at the frequency specified in the Surveillance Frequency Control Program. Each test shall include at least one channel per function.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)(d)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Containment Pressure - High	4	2	3	1, 2, 3	2
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	2
d. Automatic Actuation Logic	2	1	2	1, 2, 3	5
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Containment Pressure-- High - High	4	2(b)	3	1, 2, 3	2
c. Automatic Actuation Logic	2	1	2	1, 2, 3	5
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
c. Containment Pressure - High	4	2	3	1, 2, 3	2
d. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	2
e. Automatic Actuation Logic	2	1	2	1, 2, 3	5

TABLE 3.(Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION					
a. Manual MSI (Trip Buttons) High	2	1	2	1, 2, 3, 4	1
b. Containment Pressure-High	4	2	3	1, 2, 3	2
c. Steam Generator Pressure - Low	4	2	3	1, 2, 3(c)	2
d. Automatic Actuation Logic	2/Steam Generator	1/Steam Generator	2/Steam Generator	1, 2, 3	5
5. ENCLOSURE BUILDING FILTRATION (EBFAS)					
a. Manual EBFAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
c. Containment Pressure-High	4	2	3	1, 2, 3	2
d. Pressurizer Pressure-Low	4	2	3	1, 2, 3(a)	2
e. Automatic Actuation Logic	2	1	2	1, 2, 3	5
6. CONTAINMENT SUMP RECIRCULATION (SRAS)					
a. Manual SRAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	4
c. Automatic Actuation Logic	2	1	2	1, 2, 3	5

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. DELETED					
8. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage - level one	4/bus	2/Bus	3/bus	1, 2, 3	2
b. 4.16 kv Emergency Bus Undervoltage - level two	4/Bus	2/Bus	3/Bus	1, 2, 3	2

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. AUXILIARY FEEDWATER					
a. Manual	1/pump	1/pump	1/pump	1, 2, 3	6
b. Steam Generator Level - Low	4	2	3	1, 2, 3	2
c. Automatic Actuation Logic	2/Steam Generator	1/Steam Generator	2/Steam Generator	1, 2, 3	5
10. STEAM GENERATOR BLOWDOWN					
a. Steam Generator Level - Low	4	2	3	1, 2, 3	2

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed when pressurizer pressure is < 1850 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1850 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed when steam generator pressure is < 700 psia; bypass shall be automatically removed when steam generator pressure is ≥ 700 psia.

(d) In MODE 4 the HPSI pumps are not required to start automatically on a SIAS.

(e) DELETED

ACTION STATEMENTS

ACTION 1 - 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 COLD SHUTDOWN within the next 36 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. The inoperable channel shall either be restored to OPERABLE status, or placed in the tripped condition, within 48 hours.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also declared inoperable, and the appropriate actions are taken for the affected functional units.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

TABLE 3.3-3 (Continued)

ACTION 3 - DELETED

ACTION 4 - With the number of OPERABLE channels one less than the Total Number of Channels and with the pressurizer pressure:

a. < 1850 psia: immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing the pressurizer pressure above 1850 psia.

b. $\geq$ 1850 psia, operation may continue with the inoperable channel in the bypassed condition, provided the following condition is satisfied:

1. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.3.2.1.1 provided BOTH of the inoperable channels are placed in the bypassed condition.

ACTION 5 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
c. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 9.48 psig	≤ 10.11 psig
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
d. Pressurizer Pressure - Low	≥ 1714 psia.	≥ 1704 psia
e. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM LINE ISOLATION		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
c. Steam Generator Pressure - Low	≥ 572 psia	≥ 558 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. ENCLOSURE BUILDING FILTRATION (EBFAS)		
a. Manual EBFAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
d. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
e. Automatic Actuation Logic	Not Applicable	Not Applicable
6. CONTAINMENT SUMP RECIRCULATION (SRAS)		
a. Manual SRAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank -Low	46 ± 3 inches above tank bottom	46 ± 6 inches above tank bottom
c. Automatic Actuation Logic	Not Applicable	Not Applicable
7. DELETED		

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage - level one	≥ 2912 volts with a 2.0 ± 0.1 second time delay	≥ 2877 volts with a 2.0 ± 0.1 second time delay
b. 4.16 kv Emergency Bus Undervoltage - level two	≥ 3700 volts with an 8.0 ± 2.0 second time delay	≥ 3663 volts with an 8.0 ± 2.0 second time delay
9. AUXILIARY FEEDWATER		
a. Manual	Not Applicable	Not Applicable
b. Steam Generator Level - Low	≥ 26.8%	≥ 25.2%
c. Automatic Actuation Logic	Not Applicable	Not Applicable
10. STEAM GENERATOR BLOWDOWN		
a. Steam Generator Level - Low	≥ 26.8%	≥ 25.2%

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT			CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	SAFETY INJECTION (SIAS)					
	a.	Manual (Trip Buttons)	N.A.	N.A.	SFCP	N. A.
	b.	Containment Pressure - High	SFCP	SFCP	SFCP	1, 2, 3
	c.	Pressurizer Pressure - Low	SFCP	SFCP	SFCP	1, 2, 3
	d.	Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
2.	CONTAINMENT SPRAY (CSAS)					
	a.	Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
	b.	Containment Pressure-- High - High	SFCP	SFCP	SFCP	1, 2, 3
	c.	Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
3.	CONTAINMENT ISOLATION (CIAS)					
	a.	Manual CIAS (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
	b.	Manual SIAS (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
	c.	Containment Pressure - High	SFCP	SFCP	SFCP	1, 2, 3
	d.	Pressurizer Pressure - Low	SFCP	SFCP	SFCP	1, 2, 3
	e.	Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
4.	MAIN STEAM LINE ISOLATION					
	a.	Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
	b.	Containment Pressure - High	SFCP	SFCP	SFCP	1, 2, 3
	c.	Steam Generator Pressure - Low	SFCP	SFCP	SFCP	1, 2, 3
	d.	Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
5.	ENCLOSURE BUILDING FILTRATION (EBFAS)					
	a.	Manual EBFAS (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
	b.	Manual SIAS (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
	c.	Containment Pressure - High	SFCP	SFCP	SFCP	1, 2, 3
	d.	Pressurizer Pressure - Low	SFCP	SFCP	SFCP	1, 2, 3
	e.	Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. CONTAINMENT SUMP RECIRCULATION (SRAS)				
a. Manual SRAS (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Refueling Water Storage Tank - Low	SFCP	SFCP	SFCP	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
7. DELETED				
8. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage - level one	SFCP	SFCP	SFCP	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage - level two	SFCP	SFCP	SFCP	1, 2, 3
9. AUXILIARY FEEDWATER				
a. Manual	N.A.	N.A.	SFCP	N.A.
b. Steam Generator Level - Low	SFCP	SFCP	SFCP	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP	1, 2, 3
10. STEAM GENERATOR BLOWDOWN				
a. Steam Generator Level - Low	SFCP	SFCP	SFCP	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The coincident logic circuits shall be tested automatically or manually at the frequency specified in the Surveillance Frequency Control Program. The automatic test feature shall be verified OPERABLE at the frequency specified in the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or other specified conditions for surveillance testing of the following:

a. Pressurizer Pressure Safety Injection Automatic Actuation Logic; and

b. Pressurizer Pressure Containment Isolation Automatic Actuation Logic; and

c. Steam Generator Pressure Main Steam Line Isolation Automatic Actuation Logic; and

d. Pressurizer Pressure Enclosure Building Filtration Automatic Actuation Logic.

Testing of the automatic actuation logic for Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, and Pressurizer Pressure Enclosure Building Filtration shall be performed within 12 hours after exceeding a pressurizer pressure of 1850 psia in MODE 3. Testing of the automatic actuation logic for Steam Generator Pressure Main Steam Line Isolation shall be performed within 12 hours after exceeding a steam generator pressure of 700 psia in MODE 3.

INSTRUMENTATION

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM SENSOR CABINET POWER SUPPLY DRAWERS

LIMITING CONDITION FOR OPERATION

3.3.2.2 The engineered safety feature actuation system Sensor Cabinets (RC02A1, RC02B2, RC02C3 & RC02D4) Power Supply Drawers shall be OPERABLE and energized from the normal power source with the backup power source available. The normal and backup power sources for each sensor cabinet is detailed in Table 3.3-5a:

CABINET	NORMAL POWER	BACKUP POWER
RC02A1	VA-10	VA-40
RC02B2	VA-20	VA-30
RC02C3	VA-30	VA-20
RC02D4	VA-40	VA-10

Table 3.3-5a

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With any of the Sensor Cabinet Power Supply Drawers inoperable, or either the normal or backup power source not available as delineated in Table 3.3-5a, restore the inoperable Sensor Cabinet Power Supply Drawer to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.3.2.2.1 The engineered safety feature actuation system Sensor Cabinet Power Supply Drawers shall be determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by visual inspection of the power supply drawer indicating lamps.

4.3.2.2.2 Verify the OPERABILITY of the Sensor Cabinet Power Supply auctioneering circuit at the frequency specified in the Surveillance Frequency Control Program.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 hours or declare the channel inoperable.

b. With the number of OPERABLE channels less than the number of MINIMUM CHANNELS OPERABLE in Table 3.3-6, take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1.1 Each required radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-3.

4.3.3.1.2 DELETED

4.3.3.1.3 Verify the response time of the control room isolation channel at the frequency specified in the Surveillance Frequency Control Program.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT			MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1.	AREA MONITORS						
	a.	Deleted					
	b.	Control Room Isolation	2	ALL MODES	2 mR/hr	10^{-1} - 10^{4} mR/hr	16
	c.	Containment High Range	1	1, 2, 3, & 4	100 R/hr	10^{0} - 10^{8} R/hr	17
2.	PROCESS MONITORS						
	a.	Containment Atmosphere-Particulate	1	1, 2, 3, & 4	NA	10 - 10^{+6} cpm	14
	b.	Deleted					
	c.	Noble Gas Effluent Monitor (high range) (Unit 2 stack)	1	1, 2, 3, & 4	2 x 10^{-1} uci/cc	10^{-3} - 10^{5} uci/cc	17

TABLE NOTATION

(a) DELETED

ACTION 13 - DELETED

ACTION 14 - With the number of process monitors OPERABLE less than required by the MINIMUM CHANNELS OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 15 - DELETED

ACTION 16 - 1) With the number of OPERABLE channels one less than required by the MINIMUM CHANNELS OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days or initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

2) With the number of OPERABLE channels two less than required by the MINIMUM CHANNELS OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 17 - With the number of OPERABLE channels less than required by the MINIMUM CHANNELS OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the discovery or

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

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT			CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	AREA MONITORS					
	a.	Deleted				
	b.	Control Room Isolation	SFCP	SFCP	SFCP	ALL MODES
	c.	Containment High Range	SFCP	SFCP*	SFCP	1, 2, 3, & 4
2.	PROCESS MONITORS					
	a.	Containment Atmosphere-Particulate	SFCP	SFCP	SFCP	1, 2, 3, & 4
	b.	Deleted				
	c.	Noble Gas Effluent Monitor (high range) (Unit 2 Stack)	SFCP	SFCP	SFCP	1, 2, 3, & 4

* Calibration of the sensor with a radioactive source need only be performed on the lowest range. Higher ranges may be calibrated electronically.

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3-9, either:

a. Restore the inoperable channel to OPERABLE status within 7 days, or

b. Be in HOT SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5 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.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Wide Range Logarithmic Neutron Flux Monitor	Hot Shutdown Panel (C-21)	10^{-8}% - 100%	1
2. Reactor Trip Breaker Indication	Reactor Trip Switchgear (Q03)	OPEN-CLOSE	1/trip breaker
3. Reactor Cold Leg Temperature	Hot Shutdown Panel (C-21)	0-600°F	1
4. Pressurizer Pressure	Hot Shutdown Panel (C-21)		
a. Low Range		0-1600 psia	1
b. High Range		1500-2500 psia	1
5. Pressurizer Level	Hot Shutdown Panel (C-21)	0-100%	1
6. Steam Generator Pressure	Hot Shutdown Panel (C-21)	0-1200 psia	1/steam generator
7. Steam Generator Level	Hot Shutdown Panel (C-21	0-100%	1/steam generator

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT			CHANNEL CHECK	CHANNEL CALIBRATION
1.	Wide Range Logarithmic Neutron Flux		SFCP	SFCP*
2.	Reactor Trip Breaker Indication		SFCP	N.A.
3.	Reactor Cold Leg Temperature		SFCP	SFCP
4.	Pressurizer Pressure			
	a.	Low Range	SFCP	SFCP
	b.	High Range	SFCP	SFCP
5.	Pressurizer Level		SFCP	SFCP
6.	Steam Generator Level		SFCP	SFCP
7.	Steam Generator Pressure		SFCP	SFCP

* Neutron detectors are excluded from the CHANNEL CALIBRATION.

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. ACTIONS per Table 3.3-11.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-11
ACCIDENT MONITORING INSTRUMENTATION

Instrument		Total No. of Channels	Minimum Channels OPERABLE	ACTION
1.	Pressurizer Water Level	2	1	1
2.	Auxiliary Feedwater Flow Rate	2/S.G.	1/S.G.	1
3.	RCS Subcooled/Superheat Monitor	2	1	2
4.	PORV Position Indicator	1/valve	1/valve	3
5.	PORV Block Valve Position Indicator	1/valve	1/valve	3
6.	Safety Valve Position Indicator	1/valve	1/valve	3
7.	Containment Pressure (Wide Range)	2	1	4
8.	Containment Water Level (Narrow Range)	1	1	7##
9.	Containment Water Level (Wide Range)	2	1	4
10.	Core Exit Thermocouples	4 CETs/core quadrant	2 CETs in any of 2 core quadrants	5
11.	Main Steam Line Radiation Monitor	3	3	6
12.	Reactor Vessel Coolant Level	2*	1*	8

* A channel is eight (8) sensors in a probe. A channel is OPERABLE if four (4) or more sensors, two (2) or more in the upper four and two (2) or more in the lower four, are OPERABLE.

Refer to ACTION statement in Technical Specification 3.4.6.1.

TABLE 3.3-11 (Continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

ACTION 2 - With the number of channels OPERABLE less than the MINIMUM CHANNELS OPERABLE, determine the subcooling margin once per 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information, and monitor discharge pipe temperature once per shift to determine valve position. With the number of OPERABLE accident monitoring instrumentation channels less than the required Minimum Channels OPERABLE in Table 3.3-11 and one or more of the above mentioned quench tank parameters or discharge pipe temperature unavailable, either restore the inoperable accident monitoring instrumentation channel to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1) Initiate an alternate method for determining if there is loss of coolant through an inadvertently open valve; and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the actions taken, the cause of the malfunction, the plans for restoring the accident monitoring instrumentation channel to OPERABLE status; and

3) Restore the accident monitoring instrumentation channel to OPERABLE status by the end of the next scheduled refueling outage.

This ACTION is not required if the PORV block valve is closed with power removed in accordance with Specification 3.4.3.b or 3.4.3.c.

ACTION 4 - a. With the number of OPERABLE accident monitoring instrumentation channels less than the total number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in effect for estimating the applicable parameter during the interim.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in effect for estimating the applicable parameter during the interim.

ACTION 5 - With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours, or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 6 - With any channel of radiation monitoring instrumentation inoperable, portable hand-held radiation detection equipment will be used to assess radiation releases from the atmospheric dump valves and steam generator safeties subsequent to a steam generator tube rupture.

ACTION 7 - Restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours. (See the ACTION statement in Technical Specification 3.4.6.1.).

ACTION 8 - With the number of OPERABLE Channels one less than the MINIMUM CHANNELS OPERABLE in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. 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 system to OPERABLE status; and

2. Restore the system to OPERABLE status at the next scheduled refueling; and

3. Initiate an alternate method of monitoring the Reactor Vessel inventory.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Pressurizer Water Level	SFCP	SFCP
2.	Auxiliary Feedwater Flow Rate	SFCP	SFCP
3.	Reactor Coolant System Subcooled/Superheat Monitor	SFCP	SFCP
4.	PORV Position Indicator	SFCP	SFCP
5.	PORV Block Valve Position Indicator	N.A.	SFCP
6.	Safety Valve Position Indicator	SFCP	SFCP
7.	Containment Pressure	SFCP	SFCP
8.	Containment Water Level (Narrow Range)	SFCP	SFCP
9.	Containment Water Level (Wide Range)	SFCP	SFCP
10.	Core Exit Thermocouples	SFCP	SFCP*
11.	Main Steam Line Radiation Monitor	SFCP	SFCP
12.	Reactor Vessel Coolant Level	SFCP	SFCP*

* Electronic calibration from the ICC cabinets only.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the requirements of the above specification not met, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 Two reactor coolant loops shall be OPERABLE and one reactor coolant loop shall be in operation.

> NOTE
>
> All reactor coolant pumps may not be in operation for up to 1 hour per 8 hour period provided:
>
> a. 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; and
>
> b. core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 3.

ACTION:

a. With one reactor coolant loop inoperable, restore the required reactor coolant loop to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop OPERABLE or in operation, immediately 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 and immediately initiate corrective action to return one required reactor coolant loop to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

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

4.4.1.2.2 One reactor coolant loop shall be verified to be in operation at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.2.3 Each steam generator secondary side water level shall be verified to be ≥ 10% narrow range at the frequency specified in the Surveillance Frequency Control Program.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 Two loops or trains consisting of any combination of reactor coolant loops or shutdown cooling trains shall be OPERABLE and one loop or train shall be in operation.

NOTES

1. All reactor coolant pumps and shutdown cooling pumps may not be in operation for up to 1 hour per 8 hour period provided:
   a. 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; and
   b. core outlet temperature is maintained at least 10°F below saturation temperature.
2. The following restrictions apply when starting the first reactor coolant pump and any RCS cold leg temperature is ≤ 275°F. The first reactor coolant pump shall not be started unless:
   a. pressurizer water level is < 43.7%;
   b. pressurizer pressure is < 340 psia; and
   c. secondary water temperature in each steam generator is < 50°F above each RCS cold leg temperature.

APPLICABILITY: MODE 4

ACTION: a. With one reactor coolant loop AND two shutdown cooling trains inoperable:

Immediately initiate action to restore a second reactor coolant loop, or one shutdown cooling train to OPERABLE status.

b. With two reactor coolant loops AND one shutdown cooling train inoperable:

Immediately initiate action to restore a second shutdown cooling train, or one reactor coolant loop to OPERABLE status, and be in COLD SHUTDOWN within 24 hours.

c. With all reactor coolant loops AND shutdown cooling trains inoperable, OR no reactor coolant loop or shutdown cooling train in operation:

Immediately 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 and immediately initiate action to restore one reactor coolant loop or one shutdown cooling train to OPERABLE status and operation.

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 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 available.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE, by verifying the secondary side water level to be ≥ 10% narrow range at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.3.3 One reactor coolant loop or shutdown cooling train shall be verified to be in operation at the frequency specified in the Surveillance Frequency Control Program.

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

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

- -

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

COLD SHUTDOWN - REACTOR COOLANT SYSTEM LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4 One shutdown cooling train shall be OPERABLE and in operation, and either:

a. One additional shutdown cooling train shall be OPERABLE;

OR

b. The secondary side water level of each steam generator shall be ≥ 10% narrow range.

NOTES

1. The normal or emergency power source may be inoperable in MODE 5.

2. All shutdown cooling pumps may not be in operation for up to 1 hour per 8 hour period provided:

   a. 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; and

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

3. The following restrictions apply when starting the first reactor coolant pump and any RCS cold leg temperature is ≤ 275°F. The first reactor coolant pump shall not be started unless:

   a. pressurizer water level is < 43.7%;

   b. pressurizer pressure is < 340 psia; and

   c. secondary water temperature in each steam generator is < 50°F above each RCS cold leg temperature.

4. One required shutdown cooling train may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling train is OPERABLE and in operation.

5. All shutdown cooling trains may not be in operation during planned heatup to MODE 4 when at least one reactor coolant loop is in operation.

LIMITING CONDITION FOR OPERATION (continued)

APPLICABILITY: MODE 5 with Reactor Coolant System loops filled.

ACTION:

a. With one shutdown cooling train inoperable and any steam generator secondary water level not within limits, immediately initiate action to either restore a second shutdown cooling train to OPERABLE status or restore steam generator secondary water levels to within limit.

b. With no shutdown cooling train OPERABLE or in operation, immediately 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 and immediately initiate action to restore one shutdown cooling train to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1 The required shutdown cooling 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 available.

4.4.1.4.2 The required steam generators shall be determined OPERABLE, by verifying the secondary side water level to be ≥ 10% narrow range at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.4.3 One shutdown cooling train shall be verified to be in operation at the frequency specified in the Surveillance Frequency Control Program.

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

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - REACTOR COOLANT SYSTEM LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.5 Two shutdown cooling trains shall be OPERABLE and one shutdown cooling train shall be in operation.

NOTES

1. The normal or emergency power source may be inoperable in MODE 5.
2. All shutdown cooling pumps may not be in operation for up to 15 minutes when switching from one train to another provided:
   a. 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;
   b. core outlet temperature is maintained at least 10°F below saturation temperature; and
   c. no draining operations to further reduce Reactor Coolant System water volume are permitted.
3. The following restrictions apply when starting the first reactor coolant pump and any RCS cold leg temperature is ≤ 275°F. The first reactor coolant pump shall not be started unless:
   a. pressurizer water level is < 43.7%;
   b. pressurizer pressure is < 340 psia; and
   c. secondary water temperature in each steam generator is < 50°F above each RCS cold leg temperature
4. One shutdown cooling train may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling train is OPERABLE and in operation.

APPLICABILITY: MODE 5 with Reactor Coolant System loops not filled.

ACTION: a. With one shutdown cooling train inoperable, immediately initiate action to restore the required shutdown cooling train to OPERABLE status.

b. With no shutdown cooling train OPERABLE or in operation, immediately 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 and immediately initiate action to restore one shutdown cooling train to OPERABLE status and operation.

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - REACTOR COOLANT SYSTEM LOOPS NOT FILLED

SURVEILLANCE REQUIREMENTS

4.4.1.5.1 The required shutdown cooling 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 available.

4.4.1.5.2 One shutdown cooling train shall be verified to be in operation at the frequency specified in the Surveillance Frequency Control Program.

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

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - REACTOR COOLANT PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.6 A maximum of two reactor coolant pumps shall be OPERABLE.

APPLICABILITY: MODE 5

ACTION:

With more than two reactor coolant pumps OPERABLE, take immediate action to comply with Specification 3.4.1.6.

SURVEILLANCE REQUIREMENTS

4.4.1.6 Two reactor coolant pumps shall be demonstrated inoperable at the frequency specified in the Surveillance Frequency Control Program by verifying that the motor circuit breakers have been disconnected from their electrical power supply circuits.

REACTOR COOLANT SYSTEM

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, 3,
and 4 with all RCS cold leg temperatures > 275°F

ACTION:

With one inoperable pressurizer code safety valve, 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 pressurizer code safety valves are inoperable, be in MODE 3 within 6 hours and in MODE 4 with any RCS cold leg temperature ≤ 275°F within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 Each pressurizer code safety valve shall be demonstrated OPERABLE with a lift setting of 2500 PSIA ± 1%, in accordance with Specification 4.0.5.

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

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

Inoperable Equipment		Required ACTION	
a.	One or both PORVs, capable of being manually cycled.	a.1	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.	One PORV, not capable of being manually cycled.	b.1	Within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
c.	- - - - NOTE - - - - Not applicable when a second PORV intentionally made inoperable. - - - - - - - - - - - Two PORVs, not capable of being manual cycled.	c.1	Close the associated block valves within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
		AND	
		c.2	Remove power from associated block valves within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
		AND	

* The block valve(s) may be stroked, as necessary, during plant cooldown to prevent thermal binding.

LIMITING CONDITION FOR OPERATION

ACTION: (continued)

Inoperable Equipment	Required ACTION
c. (continued)	c.3 Verify LCO 3.7.1.2, "Auxiliary Feedwater Pumps," is met within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. AND c.4 Restore at least one PORV to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
d. One block valve.	d.1 Prevent its associated PORV from opening automatically within 1 hour, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. AND d.2 Restore the block valve to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
e. - - - - NOTE - - - - Not applicable when second block valve intentionally made inoperable. - - - - - - - - - - - Two block valves.	e.1 Verify LCO 3.7.1.2, "Auxiliary Feedwater Pumps," is met within 1 hour; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. AND e.2 Restore at least one block valve to OPERABLE status within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Revised by NRC correction letter dated January 15, 2016

SURVEILLANCE REQUIREMENTS

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

a. At the frequency specified in the Surveillance Frequency Control Program by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At the frequency specified in the Surveillance Frequency Control Program by performance of a CHANNEL CALIBRATION.

c. At the frequency specified in the Surveillance Frequency Control Program by operating the PORV through one complete cycle of full travel at conditions representative of MODES 3 or 4.

4.4.3.2 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. This demonstration is not required if a PORV block valve is closed in accordance with the ACTIONS of Specification 3.4.3.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 70%, and

b. At least two groups of pressurizer heaters each having a capacity of at least 130 kW.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Inoperable Equipment		Required ACTION	
a.	Pressurizer water level not within limit.	a.1	Be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.
b.	One group of pressurizer heaters.	b.1	Restore the inoperable group of pressurizer heaters 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 12 hours.
c.	- - - - NOTE - - - - Not applicable when second group of required pressurizer heaters intentionally made inoperable. - - - - - - - - - - - Two groups of pressurizer heaters.	c.1	Restore at least one group of pressurizer heaters to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.4.2 Verify at least two groups of pressurizer heaters each have a capacity of at least 130 kW at the frequency specified in the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

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.

STEAM GENERATOR TUBE INTEGRITY

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

<u>REMOVE THIS PAGE</u>

REMOVE THIS PAGE

<u>REMOVE THIS PAGE</u>

REMOVE THIS PAGE

REMOVE THIS PAGE

REMOVE THIS PAGE

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. One of two containment atmosphere particulate radioactivity monitoring channels, and

b. The containment sump level monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With both of the containment atmosphere particulate radioactivity monitoring channels inoperable, operation 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 COLD SHUTDOWN within the next 36 hours.

b. With the containment sump level 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 COLD SHUTDOWN within the next 36 hours.

c. With both the containment atmosphere particulate radioactivity monitoring channels inoperable and the containment sump level monitoring system inoperable, operation may continue for up to 72 hours provided:

   1. Immediate action is initiated to restore either a containment atmosphere particulate radioactivity monitoring channel or the containment sump level monitoring system to OPERABLE status, and

REACTOR COOLANT SYSTEM

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 COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment sump level monitoring system-performance of CHANNEL CALIBRATION TEST at the frequency specified in the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM 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. 75 GPD primary to secondary LEAKAGE through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE.

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 value, closed and de-activated automatic valve, blind flange, or check valve within 4 hours.

b. With any RCS operational LEAKAGE not within limits for reasons other than PRESSURE BOUNDARY LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within limits within 4 hours.

c. With ACTION and associated completion time not met, or primary to secondary LEAKAGE not within limits, be in HOT STANDBY within 6 hours and be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1

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

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

2. Not applicable to primary to secondary LEAKAGE.

- -

Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance at the frequency specified in the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.2

\- - - - - - - - - - - - - - - - - - NOTE - - - - - - - - - - - - - - - - - -

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

\- -

Verify primary to secondary LEAKAGE is ≤ 75 gallons per day through any one SG at the frequency specified in the Surveillance Frequency Control Program.

This page intentionally left blank

This page intentionally left blank

LIMITING CONDITION FOR OPERATION

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

a. ≤ 0.5 µCi/gram DOSE EQUIVALENT I-131, and

b. ≤ 550 µCi/gram DOSE EQUIVALENT XE-133.

APPLICABILITY: MODES 1, 2, 3, 4.

ACTION:

a. With the specific activity of the primary coolant > 0.5 µCi/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 ≤ 30 µCi/gram once per 4 hours.

b. With the specific activity of the primary coolant > 0.5 µCi/gram DOSE EQUIVALENT I-131 but ≤ 30 µCi/gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 0.5 µCi/gram limit. Specification 3.0.4.c is applicable.

c. With the specific activity of the primary coolant > 0.5 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or > 30 µCi/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 primary coolant > 550 µCi/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 550 µCi/gram limit. Specification 3.0.4.c is applicable.

e. With the specific activity of the primary coolant > 550 µCi/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 primary coolant ≤ 550 μCi/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 primary coolant ≤ 0.5 μCi/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 ≥ 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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 Reactor Coolant System (except the pressurizer) temperature, pressure, and heatup and cooldown rates shall be limited in accordance with the limits specified in Table 3.4-2 and shown on Figures 3.4-2a and 3.4-2b.

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 300 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.

4.4.9.1

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

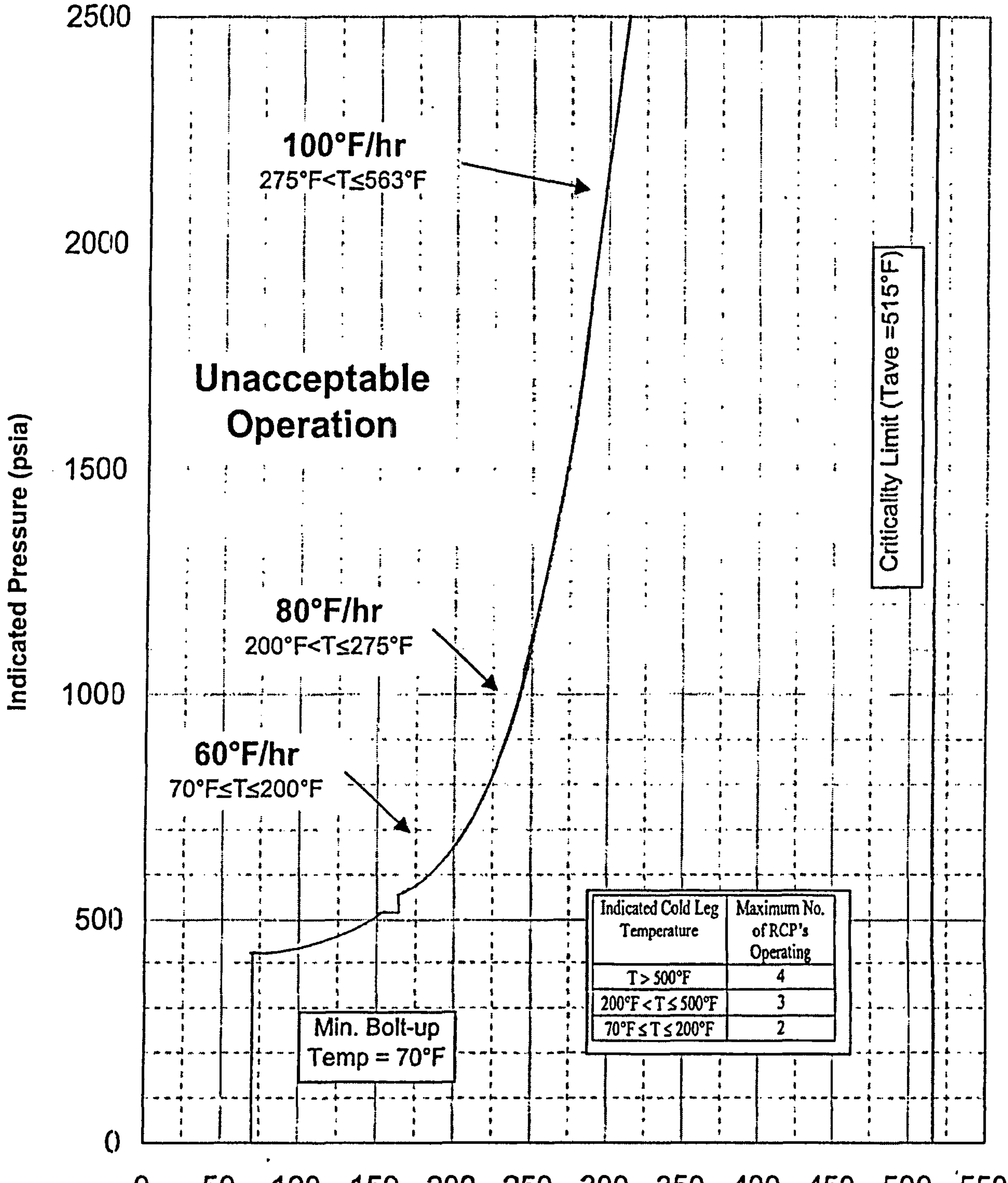
b. DELETED

TABLE 3.4-2

REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMITS

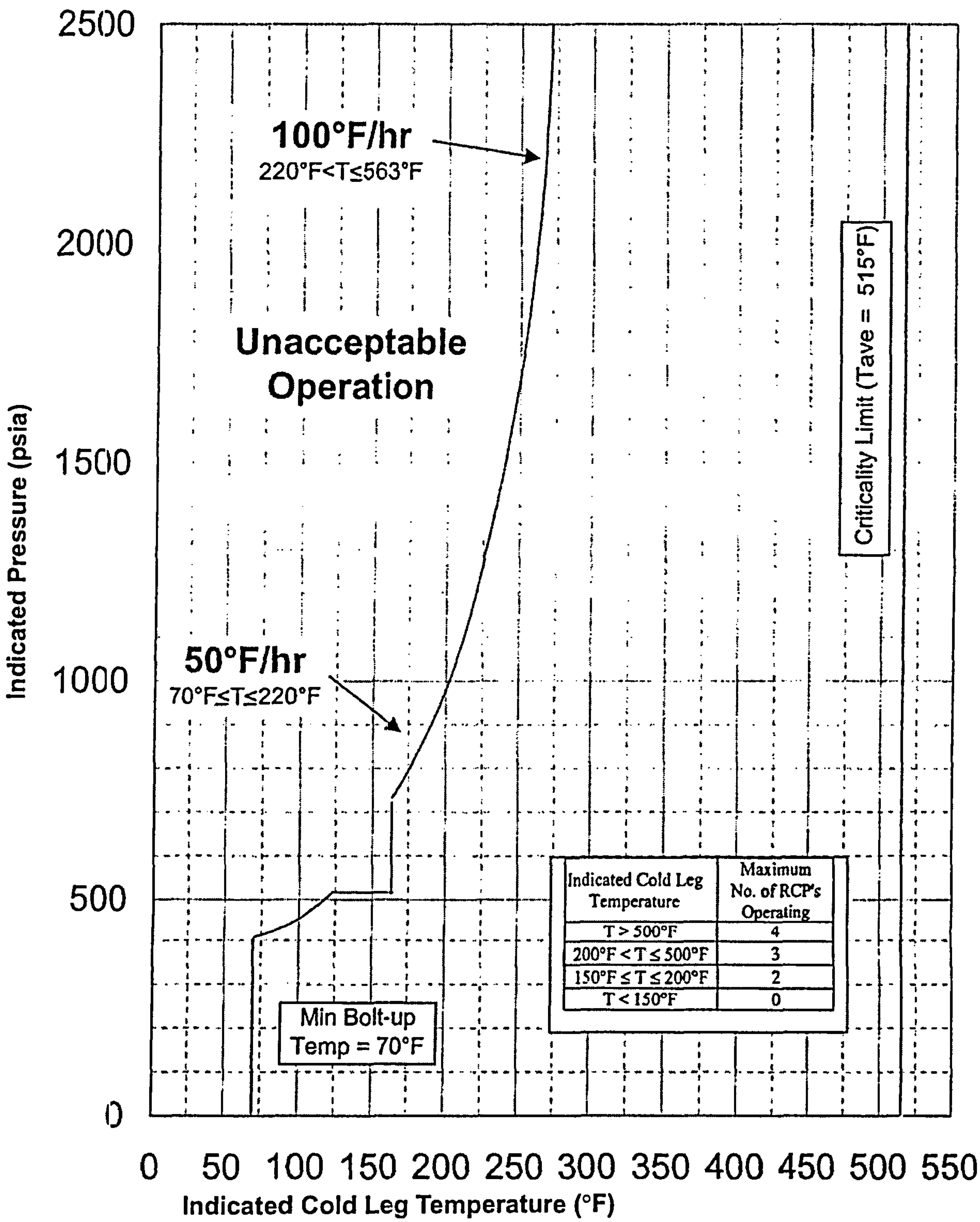
Cooldown		Heatup*	
Indicated Cold Leg Temperature	Limit	Indicated Cold Leg Temperature	Limit
≤ 220°F	≤ 50°F/hour	≤ 200°F	≤ 60°F/hour
> 220°F	≤ 100°F/hour	200°<T≤275°F	≤ 80°F/hour
		> 275°F	≤ 100°F/hour

* These limitations also apply to hydrostatic and leak test conditions.



Indicated Cold Leg Temperature	Maximum No. of RCP's Operating
T > 500°F	4
200°F < T ≤ 500°F	3
70°F ≤ T ≤ 200°F	2

Millstone Unit 2 Reactor Coolant System
Heatup Limitations for Up to 54 EFPY
Figure 3.4-2a



Millstone Unit 2 Reactor Coolant System
Cooldown Limitations for Up to 54 EFPY
Figure 3.4-2b

THIS PAGE INTENTIONALLY LEFT BLANK

This page intentionally left blank

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 A Low Temperature Overpressure Protection (LTOP) System, as specified below, shall be OPERABLE.

a. MODE 4, and MODE 5 with all RCS cold leg temperature > 190°F:

1. Maximum of two charging pumps and one HPSI pump may be capable of injecting into the RCS; and

2. Two OPERABLE PORVs with a lift setpoint of ≤ 415 psia.

b. MODE 5 with any RCS cold leg temperature ≤ 190 °F, and MODE 6 either:

1. Maximum of one charging pump may be capable of injecting into the RCS; and

2. Two OPERABLE PORVs with a lift setpoint of ≤ 415 psia.

OR

3. Maximum of two charging pumps and one HPSI pump may be capable of injecting into the RCS; and

4. The RCS is depressurized and an RCS vent of ≥ 2.2 sq. inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 275°F, MODE 5, and MODE 6 when the head is on the reactor vessel.

ACTION:

NOTE

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

a. With one required PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a ≥ 2.2 square inch vent within the next 8 hours.

b. With one required PORV inoperable in MODES 5 or 6, either restore inoperable PORV to OPERABLE status within 24 hours or depressurize and vent the RCS through a ≥ 2.2 square inch vent within the next 8 hours.

c. With both required PORVs inoperable, depressurize and vent the RCS through a ≥ 2.2 square inch vent within 8 hours.

d. With more than the maximum allowed pumps capable of injecting into the RCS, take immediate action to comply with 3.4.9.3.

e. In the event either the PORVs or the RCS vent(s) 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 or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL 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.

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

d. Testing in accordance with the inservice test requirements of Specification 4.0.5.

4.4.9.3.2 Verify no more than the maximum allowed number of charging pumps are capable of injecting into the RCS at the frequency specified in the Surveillance Frequency Control Program.

4.4.9.3.3 Verify no more than the maximum allowed number of HPSI pumps are capable of injecting into the RCS at the frequency specified in the Surveillance Frequency Control Program.

4.4.9.3.4 Verify the required RCS vent is open at the frequency specified in the Surveillance Frequency Control Program when the vent pathway is provided by vent valve(s) that is(are) locked, sealed, or otherwise secured in the open position, otherwise, verify the vent pathway at the frequency specified in the Surveillance Frequency Control Program.

This page intentionally left blank

This page intentionally left blank

This page intentionally left blank

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS (SITs)

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system SIT shall be OPERABLE with:

a. The isolation valve open and the power to the valve operator removed,

b. Between 1080 and 1190 cubic feet of borated water,

c. A minimum boron concentration of 1720 PPM, and

d. A nitrogen cover-pressure of between 200 and 250 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one SIT inoperable due to boron concentration not within limits, restore boron concentration to within limits within 72 hours.

b. With one SIT inoperable due solely to inability to verify level or pressure, restore SIT to OPERABLE status within 72 hours.

c. With one SIT inoperable, except as a result of boron concentration not within limits or inoperable level or pressure instrumentation, restore SIT to OPERABLE status within 24 hours.

d. With required ACTION a. or b. or c. and associated Completion Time not met:

   1. Be in MODE 3 within 6 hours, and

   2. Reduce pressurizer pressure to < 1750 psia within 12 hours.

e. With two or more SITs inoperable, immediately enter LCO 3.0.3.

---

*With pressurizer pressure $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS

4.5.1 Each SIT shall be demonstrated OPERABLE:

a. Verify each SIT isolation valve is fully open at the frequency specified in the Surveillance Frequency Control Program.*(1)

b. Verify borated water volume in each SIT is ≥ 1080 cubic feet and ≤ 1190 cubic feet at the frequency specified in the Surveillance Frequency Control Program.**(2)

c. Verify nitrogen cover-pressure in each SIT is ≥ 200 psig and ≤ 250 psig at the frequency specified in the Surveillance Frequency Control Program.***(3)

d. Verify boron concentration in each SIT is ≥ 1720 ppm at the frequency specified in the Surveillance Frequency Control Program, and once within 6 hours after each solution volume increase of ≥ 1% of tank volume****(4) that is not the result of addition from the refueling water storage tank.

e. Verify that the closing coil in the valve breaker cubicle is removed at the frequency specified in the Surveillance Frequency Control Program.

*(1) If one SIT is inoperable, except as a result of boron concentration not within limits, or inoperable level or pressure instrumentation, surveillance is not applicable to the affected SIT.

**(2) If one SIT is inoperable due solely to inoperable water level instrumentation, surveillance is not applicable to the affected SIT.

***(3) If one SIT is inoperable due solely to inoperable pressure instrumentation, surveillance is not applicable to affected SIT.

****(4)Only required to be performed for affected SIT.

ECCS SUBSYSTEMS - $T_{avg} \geq 300°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two ECCS subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia 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.

* With pressurizer pressure ≥ 1750 psia.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

NOTE

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

a. At the frequency specified in the Surveillance Frequency Control Program by verifying each Emergency Core Cooling System manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that the following valves are in the indicated position with power to the valve operator removed:

Valve Number	Valve Function	Valve Position
2-SI-306	Shutdown Cooling Flow Control	Open*
2-SI-659	SRAS Recirc.	Open**
2-SI-660	SRAS Recirc.	Open**

* Pinned and locked at preset throttle open position.

** To be closed prior to recirculation following LOCA.

c. By verifying the developed head of each high pressure safety injection pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

d. By verifying the developed head of each low pressure safety injection pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

e. Deleted

f. At the frequency specified in the Surveillance Frequency Control Program by verifying each Emergency Core Cooling System automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

g. At the frequency specified in the Surveillance Frequency Control Program by verifying each high pressure safety injection pump and low pressure safety injection pump starts automatically on an actual or simulated actuation signal.

h. At the frequency specified in the Surveillance Frequency Control Program by verifying each low pressure safety injection pump stops automatically on an actual or simulated actuation signal.

i. By verifying the correct position of each electrical and/or mechanical position stop for each injection valve in Table 4.5-1:

   1. Within 4 hours after completion of valve operations.

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

j. At the frequency specified in the Surveillance Frequency Control Program by verifying through visual inspection of the containment sump that each Emergency Core Cooling System subsystem suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.

k. At the frequency specified in the Surveillance Frequency Control Program by verifying the Shutdown Cooling System open permissive interlock prevents the Shutdown Cooling System inlet isolation valves from being opened with an actual or simulated Reactor Coolant System pressure signal of $\geq$ 300 psia.

l. At the frequency specified in the Surveillance Frequency Control Program by verifying that ECCS locations susceptible to gas accumulation are sufficiently filled with water.

Page 3/4 5-5a has been removed from Tech Specs as a result of this LBDCR

Table 4.5-1

ECCS INJECTION VALVES

1.	2-SI-617	"A" HPSI Header - Loop 1A Injection
2.	2-SI-627	"A" HPSI Header - Loop 1B Injection
3.	2-SI-637	"A" HPSI Header - Loop 2A Injection
4.	2-SI-647	"A" HPSI Header - Loop 2B Injection
5.	2-SI-616	"B" HPSI Header - Loop 1A Injection
6.	2-SI-626	"B" HPSI Header - Loop 1B Injection
7.	2-SI-636	"B" HPSI Header - Loop 2A Injection
8.	2-SI-646	"B" HPSI Header - Loop 2B Injection
9.	2-SI-615	LPSI Header - Loop 1A Injection
10.	2-SI-625	LPSI Header - Loop 1B Injection
11.	2-SI-635	LPSI Header - Loop 2A Injection
12.	2-SI-645	LPSI Header - Loop 2B Injection

Page 3/4 5-6a has been removed from Tech Specs as a result of this LBDCR

ECCS SUBSYSTEMS - T_{avg} < 300°F

LIMITING CONDITION FOR OPERATION

3.5.3 One high pressure safety injection subsystem shall be OPERABLE.

1. The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 4 for the high pressure safety injection pump that is inoperable pursuant to Specification 3.4.9.3 provided the high pressure safety injection pump is restored to OPERABLE status within 1 hour after entering MODE 4.
2. In MODE 4, the requirement for OPERABLE safety injection and sump recirculation actuation signals is satisfied by use of the safety injection and sump recirculation trip pushbuttons.
3. In MODE 4, the OPERABLE HPSI pump is not required to start automatically on a SIAS. Therefore, the pump control switch for this OPERABLE pump may be placed in the pull-to-lock position without affecting the OPERABILITY of this pump.

APPLICABILITY: MODES 3* and 4.

ACTION:

NOTE

LCO 3.0.4.b is not applicable to ECCS high pressure safety injection subsystem when entering MODE 4

a. With no high pressure safety injection subsystem OPERABLE, restore at least one high pressure safety injection subsystem to OPERABLE status within one hour or be in COLD SHUTDOWN within the next 24 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.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The high pressure safety injection subsystem shall be demonstrated OPERABLE per the applicable portions of Surveillance Requirements 4.5.2.a, 4.5.2.b, 4.5.2.c, 4.5.2.f, 4.5.2.g, 4.5.2.i, 4.5.2.j, and 4.5.2.l.

---

* With pressurizer pressure < 1750 psia.

THIS PAGE INTENTIONALLY DELETED

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum contained volume of 370,000 gallons of borated water,

b. A minimum boron concentration of 1720 ppm,

c. A minimum water temperature of 50°F when in MODES 1 and 2, and

d. A minimum water temperature of 35°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore tank to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 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 water level in the tank, and

2. Verifying the boron concentration of the water.

b. When in MODES 3 and 4, at the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature is ≥ 35°F when the RWST ambient air temperature is < 35°F.

c. When in MODES 1 and 2, at the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature is ≥ 50°F when the RWST ambient air temperature is < 50°F.

EMERGENCY CORE COOLING SYSTEMS

TRISODIUM PHOSPHATE (TSP)

LIMITING CONDITION FOR OPERATION

3.5.5 The TSP baskets shall contain ≥282 ft^3 of active TSP.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With the quantity of TSP less than required, restore the TSP quantity within 72 hours, or be in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.5.1 Verify that the TSP baskets contain ≥282 ft^3 of TSP at the frequency specified in the Surveillance Frequency Control Program.

4.5.5.2 Verify that a sample from the TSP baskets provides adequate pH adjustment of borated water at the frequency specified in the Surveillance Frequency Control Program.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 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 3.6.3.1.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying the equipment hatch is closed and sealed.

c. By verifying the containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. After each closing of a penetration subject to type B testing (except the containment air lock), if opened following a Type A or B test, by leak rate testing in accordance with the Containment Leakage Rate Testing Program.

e. By verifying Containment structural integrity in accordance with the Containment Tendon Surveillance Program.

---

(1) Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise9 secured in the closed position. These penetrations shall be verified closed prior to entering MODE 4 from MODE 5, if not performed within the previous 92 days.

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

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

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $< L_a$, 0.50 percent by weight of the containment air per 24 hours at P_a.

b. A combined leakage rate of $< 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

c. A combined leakage rate of $< 0.014\ L_a$ for all penetrations that are secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.014 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

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

This Page Intentionally Deleted

This Page Intentionally Deleted

This Page Intentionally Deleted.

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 of $\leq 0.05\ L_a$ at P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: :

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

a. With 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 until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

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

4. Entry into an OPERATIONAL MODE or other specified condition under the provisions of Specification 3.0.4 shall not be made if the inner air lock door is inoperable.

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 there after. 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).

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

4.6.1.3.1 Each containment air lock shall be demonstrated OPERABLE 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)

4.6.1.3.2 Each containment air lock shall be demonstrated OPERABLE 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.

This Page Intentionally Deleted

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -12 inches Water Gauge and +1.0 PSIG.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure in excess of or below the limits above, restore the internal pressure to within the limits within 1 hour or be in HOT STANDBY within the next 4 hours; go to COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to within the limits at the frequency specified in the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 120°F, reduce the average air temperature to within the limit within 8 hours, or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be determined to be ≤ 120°F at the frequency specified in the Surveillance Frequency Control Program.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains, with each cooling train consisting of two containment air recirculation and cooling units, shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

Inoperable Equipment		Required ACTION	
a.	One containment spray train	a.1	Restore the inoperable containment spray train to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia within the following 6 hours.
b.	One containment cooling train	b.1	Restore the inoperable containment cooling train to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.
c.	One containment spray train AND One containment cooling train	c.1	Restore the inoperable containment spray train or the inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
d.	- - - - NOTE - - - - Not applicable when second containment spray train intentionally made inoperable. - - - - - - - - - - - Two containment spray trains.	d.1 d.2	Verify LCO 3.7.6.1, "Control Room Emergency Ventilation System," is met within 1 hour or be in HOT SHUTDOWN within the next 12 hours. Restore at least one inoperable containment spray train to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 12 hours.
e.	Two containment cooling trains	e.1	Restore at least one inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
f.	All other combinations	f.1	Enter LCO 3.0.3 immediately.

* The Containment Spray System is not required to be OPERABLE in MODE 3 if pressurizer pressure is < 1750 psia.

SURVEILLANCE REQUIREMENTS

4.6.2.1.1 Each containment spray train shall be demonstrated OPERABLE:

\- - - - - - - - - - - - - - - - - NOTE - - - - - - - - - - - - - - - - - -

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

\- -

a. At the frequency specified in the Surveillance Frequency Control Program by verifying each containment spray manual, power operated, and automatic valve in the spray train flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. By verifying the developed head of each containment spray pump at the flow test 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 each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

d. At the frequency specified in the Surveillance Frequency Control Program by verifying each containment spray pump starts automatically on an actual or simulated actuation signal.

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

f. At the frequency specified in the Surveillance Frequency Control Program by verifying the Containment Spray System locations susceptible to gas accumulation are sufficiently filled with water.

4.6.2.1.2 Each containment air recirculation and cooling unit shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by operating each containment air recirculation and cooling unit in slow speed for $\geq$ 15 minutes.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying each containment air recirculation and cooling unit cooling water flow rate is $\geq$ 500 gpm.

c. At the frequency specified in the Surveillance Frequency Control Program by verifying each containment air recirculation and cooling unit starts automatically on an actual or simulated actuation signal.

THIS PAGE
INTENTIONALLY LEFT BLANK

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.(1) (2)

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, either:

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 a deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of a 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 COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE:

a. By verifying the isolation time of each power operated automatic containment isolation valve when tested pursuant to Specification 4.0.5.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.

---

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

(2) The provisions of this Specification in MODES 1, 2 and 3, 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.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.2 The containment purge supply and exhaust isolation valves shall be sealed closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment purge supply and/or one exhaust isolation valve open, close the open valve(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.2 The containment purge supply and exhaust isolation valves shall be determined sealed closed at the frequency specified in the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

DELETED

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

This Page Intentionally Left Blank.

POST-INCIDENT RECIRCULATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.4.4 Two separate and independent post-incident recirculation systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one post-incident recirculation system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.4 Each post-incident recirculation system shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Verifying that the system can be started on operator action in the control room, and

b. Verifying that the system operates for at least 15 minutes.

3/4.6.5 SECONDARY CONTAINMENT

ENCLOSURE BUILDING FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two separate and independent Enclosure Building Filtration Trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Inoperable Equipment	Required ACTION
a. One Enclosure Building Filtration Train.	a.1 Restore the inoperable Enclosure Building Filtration Train to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.
b. – – – - NOTE - – – – Not applicable when second Enclosure Building Filtration Train intentionally made inoperable. – – – – – – – – – – – Two Enclosure Building Filtration Trains.	b.1 Verify at least one train of containment spray is OPERABLE within 1 hour or be in COLD SHUTDOWN within the next 36 hours. AND b.2 Restore at least one Enclosure Building Filtration Train to OPERABLE status within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each Enclosure Building Filtration Train shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 continuous minutes.

b. At the frequency specified in the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, and (2) following painting, fire or chemical release in any ventilation zone communicating with the train by:

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 9000 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, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a train flow rate of 9000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

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, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

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 ≤ 2.6 inches Water Gauge while operating the train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the train starts on an Enclosure Building Filtration Actuation Signal (EBFAS).

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

---

* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89. Additionally, the charcoal sample shall have a removal efficiency of ≥ 95%.

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

CONTAINMENT SYSTEMS

ENCLOSURE BUILDING

LIMITING CONDITION FOR OPERATION

3.6.5.2 The Enclosure Building shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the Enclosure Building inoperable, restore the Enclosure Building to OPERABLE status within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

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

4.6.5.2.2. At the frequency specified in the Surveillance Frequency Control Program verify each Enclosure Building Filtration Train produces a negative pressure of greater than or equal to 0.25 inches W.G. in the Enclosure Building Filtration Region within 1 minute after an Enclosure Building Filtration Actuation Signal.

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 shall be OPERABLE with lift settings as specified in Table 4.7-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more required main steam line code safety valves per steam generator inoperable,

   1. Reduce THERMAL POWER within 4 hours to less than or equal to the applicable percent of RATED THERMAL POWER listed in Table 3.7-1, and

   2. Reduce the Power Level-High trip setpoint in accordance with Table 3.7-1 within 36 hours.

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

b. With more than four main steam line code safety valves on a single steam generator inoperable, be in HOT STANDBY within 6 hours, and HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each main steam line code safety valve shall be demonstrated OPERABLE, with lift settings as shown in Table 4.7-1, in accordance with Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE MAIN STEAM LINE CODE SAFETY VALVES (MSSVs)

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM POWER (Percent Of RATED THERMAL POWER)	MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT (Percent Of RATED THERMAL POWER)
8	100	106.6 (Ceiling)
7	85	94.6
6	75	84.6
5	60	69.6
4	45	54.6

TABLE 4.7-1

STEAM LINE SAFETY VALVES

VALVE NUMBERS	LIFT SETTING* (± 3%)**
a. 2-MS-246 & 2-MS-247	1000 psia
b. 2-MS-242 & 2-MS-254	1005 psia
c. 2-MS-245 & 2-MS-249	1015 psia
d. 2-MS-241 & 2-MS-252	1025 psia
e. 2-MS-244 & 2-MS-251	1035 psia
f. 2-MS-240 & 2-MS-250	1045 psia
g. 2-MS-239, 2-MS-243, 2-MS-248 & 2-MS-253	1050 psia

* 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 PUMPS

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three steam generator auxiliary feedwater pumps shall be OPERABLE with:

a. Two feedwater pumps capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

NOTE

LCO 3.0.4.b is not applicable

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

Inoperable Equipment	Required ACTION
a. Turbine-driven auxiliary feedwater pump due to one steam supply 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.

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 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying each auxiliary feedwater manual, power operated, and automatic valve in each water flow path and in each steam supply flow path to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. By verifying 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. (Not required to be performed for the steam turbine driven auxiliary feedwater pump until 24 hours after reaching 800 psig in the steam generators. 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 each auxiliary feedwater automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position, as designed, on an actual or simulated actuation signal.

d. At the frequency specified in the Surveillance Frequency Control Program by verifying each auxiliary feedwater pump starts automatically, as designed, on an actual or simulated actuation signal.

e. By verifying the proper alignment of the required auxiliary feedwater flow paths by verifying flow from the condensate storage tank to each steam generator prior to entering MODE 2 whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of greater than 30 days.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 165,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With less than 165,000 gallons of water in the condensate storage tank, within 4 hours either:

a. Restore the water volume to within the limit or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the fire water system as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank water volume to within its limits within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying the water level.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be ≤ 0.05 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system > 0.05 μCi/gram DOSE EQUIVALENT I-131, be in COLD SHUTDOWN within 36 hours after detection.

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 of Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY
1. Gross Activity Determination	At the frequency specified in the Surveillance Frequency Control Program.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit b) At the frequency specified in the Surveillance Frequency Control Program, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in MODE 2 within the next 6 hours.

MODES 2 and 3 - With one or more main steam line isolation valves inoperable, subsequent operation in MODES 2 or 3 may continue provided the inoperable valve(s) is(are) restored to OPERABLE status or the isolation valve(s) is(are) closed* within 1 hour and verified closed at least once per 7 days; otherwise, be in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 6 seconds on any closure actuation signal while in HOT STANDBY, with $T_{avg} \geq 515°F$ during each plant startup except that verification of full closure within 6 seconds need not be determined more often than once per 92 days. The provisions of Technical Specification 4.0.4 do not apply for entry into MODE 3.

*The main steam line isolation valves may be opened to perform Surveillance Requirement 4.7.1.5.

MAIN FEEDWATER ISOLATION COMPONENTS (MFICs)

LIMITING CONDITION FOR OPERATION

3.7.1.6 Each feedwater isolation component listed in Table 3.7-3 shall be OPERABLE.

FW Isolation Components	Description
FW-38A	A FP Discharge MOV
FW-38B	B FP Discharge MOV
FW-42A	A FW Block MOV
FW-42B	B FW Block MOV
FW-41A	A FW Regulating Bypass Valve
FW-41B	B FW Regulating Bypass Valve
FW-51A	A FW Regulating Valve
FW-51B	B FW Regulating Valve
H5A	A SG Feedwater Pump Trip Circuitry
H5B	B SG Feedwater Pump Trip Circuitry

Table 3.7-3

APPLICABILITY: MODES 1, 2 & 3

ACTION:

a. With one feedwater isolation component inoperable in either or both feedwater flow paths, either:

1. Restore the inoperable component(s) to OPERABLE status within 72 hours, or

2. Close or isolate the inoperable feedwater isolation valve(s) within 72 hours, and verify that the inoperable feedwater isolation valve(s) is closed or isolated once per 7 days, or

3. Secure or isolate the feedwater pump(s) with inoperable feedwater pump trip circuitry within 72 hours and verify that the inoperable feedwater pump(s) is secured or isolated once per 7 days, or

4. Be in HOT SHUTDOWN within the next 12 hours.

MAIN FEEDWATER ISOLATION COMPONENTS (MFICs)

LIMITING CONDITION FOR OPERATION (Continued)

b. With two or more of the feedwater isolation components inoperable in the same flow path, either:

1. Restore the inoperable component(s) to OPERABLE status within 8 hours until ACTION 'a' applies, or

2. Isolate the affected flow path within 8 hours, and verify that the inoperable feedwater isolation components are closed or isolated/secured once per 7 days, or

3. Be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each feedwater isolation valve/feedwater pump trip circuitry shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Verifying that on 'A' main steam isolation test signal, each isolation valve actuates to its isolation position, and

b. Verifying that on 'B' main steam isolation test signal, each isolation valve actuates to its isolation position, and

c. Verifying that on 'A' main steam isolation test signal, each feedwater pump trip circuit actuates, and

d. Verifying that on 'B' main steam isolation test signal, each feedwater pump trip circuit actuates.

PLANT SYSTEMS

ATMOSPHERIC DUMP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.7 Each atmospheric dump valve line shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one atmospheric dump valve line inoperable, restore the inoperable line to OPERABLE status within 48 hours or be in MODE 3 within the next 6 hours and MODE 4 within the following 24 hours.

b. With more than one atmospheric dump valve line inoperable, restore one inoperable line to OPERABLE status within 1 hour or be in MODE 3 within the next 6 hours and MODE 4 within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.7 Verify the OPERABILITY of each atmospheric dump valve line by local manual operation of each valve in the flowpath through one complete cycle of operation at the frequency specified in the Surveillance Frequency Control Program.

PLANT SYSTEMS

STEAM GENERATOR BLOWDOWN ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.8 Each steam generator blowdown isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With one or more steam generator blowdown isolation valves inoperable, either:

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

b. Isolate the affected steam generator blowdown line within 4 hours; or

c. Be in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.8 Verify the closure time of each steam generator blowdown isolation valve is ≤ 10 seconds on an actual or simulated closure signal at the frequency specified in the Surveillance Frequency Control Program.

THIS PAGE INTENTIONALLY LEFT BLANK

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 Two reactor building closed cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one reactor building closed cooling water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 Each reactor building closed cooling water loop shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying each reactor building closed cooling water manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying each reactor building closed cooling water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

c. At the frequency specified in the Surveillance Frequency Control Program by verifying each reactor building closed cooling water pump starts automatically on an actual or simulated actuation signal.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 Two service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one service water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying each service water manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying each service water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

c. At the frequency specified in the Surveillance Frequency Control Program by verifying each service water pump starts automatically on an actual or simulated actuation signal.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent Control Room Emergency Ventilation Trains shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6.

During movement of recently irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

Inoperable Equipment	Required ACTION
a. One Control Room Emergency Ventilation Train, except as specified in ACTION c.	a.1 Restore the inoperable Control Room Emergency Ventilation Train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
b. - - - - NOTE - - - - Not applicable when second Control Room Emergency Ventilation Train intentionally made inoperable. - - - - - - - - - - - Two Control Room Emergency Ventilation Trains, except as specified in ACTION c.	b.1 Initiate action to implement mitigating actions immediately or be in HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours. AND b.2 Verify LCO 3.4.8, "Reactor Coolant System, Specific Activity," is met within 1 hour or be in HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours. AND b.3 Restore at least one Control Room Emergency Ventilation Train to OPERABLE status within 24 hours or be in HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

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

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

MODES 1, 2, 3, and 4:

Inoperable Equipment	Required ACTION
c. One or more Control Room Emergency Ventilation Trains, due to an inoperable CRE boundary.	c.1 Immediately initiate action to implement mitigating actions or be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours. AND c.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 or be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours. AND c.3 Restore CRE boundary to OPERABLE status within 90 days or be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours.

ACTION: (Continued)

MODES 5 and 6, and during movement of recently irradiated fuel assemblies:**

Inoperable Equipment		Required ACTION	
d.	One Control Room Emergency Ventilation Train, except due to an inoperable CRE boundary.	d.1	Restore the inoperable Control Room Emergency Ventilation Train to OPERABLE status within 7 days.
		AND	
		d.2	After 7 days, initiate and maintain operation of the remaining OPERABLE Control Room Emergency Ventilation Train in the recirculation mode of operation or immediately suspend the movement of recently irradiated fuel assemblies.
e.1	Both Control Room Emergency Ventilation Trains,	e.1	Immediately suspend the movement of recently irradiated fuel assemblies.
OR			
e.2	The OPERABLE Control Room Emergency Ventilation Train required to be in the recirculation mode by ACTION d. not capable of being powered by an OPERABLE normal and emergency power source,		
OR			
e.3	One or more Control Room Emergency Ventilation Trains, due to an inoperable CRE boundary.		

** In MODES 5 and 6, when a Control Room Emergency Ventilation Train is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of 3.7.6.1 Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system (s), subsystem (s), train (s), component (s) and device(s) are OPERABLE, or likewise satisfy the requirements of the specification. Unless both conditions (1) and (2) are satisfied within 2 hours, then ACTION 3.7.6.1.d or 3.7.6.1.e shall be invoked as applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each Control Room Emergency Ventilation Train shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that the control room air temperature is ≤ 100°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 adsorber train and verifying that the train operates for at least 15 minutes.

c. At the frequency specified in the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, and (2) following painting, fire or chemical release in any ventilation zone communicating with the train by:

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 2500 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, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.* The carbon sample shall have a removal efficiency of ≥ 95 percent.

3. Verifying a train flow rate of 2500 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

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, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

---

* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89.

SURVEILLANCE REQUIREMENTS (Continued)

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 3.4 inches Water Gauge while operating the train at a flow rate of 2500 cfm ± 10%.

2. Verifying that on a recirculation signal, with the Control Room Emergency Ventilation Train operating in the normal mode and the smoke purge mode, the train automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Deleted

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

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.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

PLANT SYSTEMS

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8 on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for the system.

SURVEILLANCE REQUIREMENTS

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

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 INTENTIONALLY LEFT BLANK

3/4.7.9 Deleted

3/4.7.10 Deleted

3/4.7.11 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.11 The ultimate heat sink 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

4.7.11 The ultimate heat sink 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.

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 a separate fuel oil supply tank containing a minimum of 12,000 gallons of fuel.

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 for remaining offsite circuit within 1 hour prior to or after entering this condition, and at least once per 8 hours thereafter. AND a.2 Restore the inoperable offsite circuit to OPERABLE status within 72 hours (within 10 days* if Required ACTION a.3 is met) or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. AND

ACTION (Continued)

Inoperable Equipment	Required ACTION
a. One offsite circuit	a.3 With MPS3 in MODE 5, 6, or defueled, the MPS3 'A' RSST inoperable, and the MPS3 'A' NSST energized with breaker 15G-13T-2 (13T) and associated disconnect switches closed, restore either offsite circuit to OPERABLE status within 10 days* if the following requirements are met: - Within 30 days prior to entering the 10-day* AOT, the availability of the supplemental power source (MPS3 SBO diesel generator) shall be verified. - During the 10-day*AOT, the availability of the supplemental power source shall be checked once per shift. If the supplemental power source becomes unavailable at any time during the 10-day*AOT, restore to available status within 24 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. - The risk management actions contained in DENC letter 20-109, Attachment 4 (also provided in TS Bases 3/4.8), shall remain in effect during the 10-day*AOT.

* To facilitate replacement of the MPS3 'A' RSST and associated equipment, use of a one-time 35-day allowed outage time is permitted provided the requirements of Required ACTION a.3 are met. The work shall be completed no later than the end of MPS3 Refueling Outage 22 (fall 2023).

ACTION (Continued)

Inoperable Equipment	Required ACTION
b. One diesel generator	b.1 Perform Surveillance Requirement 4.8.1.1.1 for the offsite circuit within 1 hour prior to or after entering this condition, and at least once per 8 hours thereafter. AND 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.2 for the OPERABLE diesel generator within 24 hours. AND b.3 Verify the steam-driven auxiliary feedwater pump is OPERABLE (MODES 1, 2, and 3 only). If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours. AND 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. 3 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. 3 diesel generator to OPERABLE status and/or Millstone Unit No. 3 SBO diesel generator to available status within 72 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. AND 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 HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

ACTION (Continued)

Inoperable Equipment	Required ACTION
c. One offsite circuit AND One diesel generator	c.1 Perform Surveillance Requirement 4.8.1.1.1 for remaining offsite circuit within 1 hour and at least once per 8 hours thereafter. AND 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.2 for the OPERABLE diesel generator within 8 hours. AND c.3 Verify the steam-driven auxiliary feedwater pump is OPERABLE (MODES 1, 2, and 3 only). If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours. AND c.4 Restore one inoperable A.C. source to OPERABLE status within 12 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. AND 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 HOT STANDBY within the next 6 hours. AND 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.

ACTION (Continued)

Inoperable Equipment	Required ACTION
e. Two diesel generators	e.1 Perform Surveillance Requirement 4.8.1.1.1 for the offsite circuits within 1 hour and at least once per 8 hours thereafter. AND e.2 Restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. AND 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 Verify correct breaker alignment and indicated power available for each required offsite circuit at the frequency specified in the Surveillance Frequency Control Program.

4.8.1.1.2 Each required 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 fuel oil supply tank,

2.

NOTES
1. A modified diesel generator start involving idling and gradual acceleration to synchronous speed may be used as recommended by the manufacturer. When modified start procedures are not used, the requirements of SR 4.8.1.1.2.d.1 must be met.
2. Performance of SR 4.8.1.1.2.d satisfies this Surveillance Requirement.

Verifying the diesel generator starts from standby conditions and achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and Frequency ≥ 58.8 Hz and ≤ 61.2 Hz.

3.

NOTES
1. Diesel generator loading may include gradual loading as recommended by the manufacturer.
2. Momentary transients outside the load range do not invalidate this test.
3. This test shall be conducted on only one diesel generator at a time.
4. This test shall be preceded by and immediately follow without shutdown a successful performance of SR 4.8.1.1.2.a.2, or SRs 4.8.1.1.2.d.1 and 4.8.1.1.2.d.2.
5. Performance of SR 4.8.1.1.2.d satisfies this Surveillance Requirement.

Verifying the diesel generator is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.

---

* All diesel starts may be preceded by an engine prelube period.

b. The diesel fuel oil supply shall be checked by:

1. Checking for and removing accumulated water from each fuel oil storage tank at the frequency specified in the Surveillance Frequency Control Program.

2. Verifying fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program in accordance with the Diesel Fuel Oil Testing Program.

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

1. Deleted

2.

NOTE

This surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.

Verifying that the automatic time delay sequencer is OPERABLE with the following settings:

Sequence Step	Time After Closing of Diesel Generator Output Breaker (Seconds)	
	Minimum	Maximum
1 (T_1)	1.5	2.2
2 (T_2)	$T_1 + 5.5$	8.4
3 (T_3)	$T_2 + 5.5$	14.6
4 (T_4)	$T_3 + 5.5$	20.8

SURVEILLANCE REQUIREMENTS (Continued)

3.

> NOTE
>
> If performed with the diesel generator synchronized with offsite power, this surveillance shall be performed at a power factor ≤ 0.9 lagging. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.

Verifying the diesel generator capability to reject a load greater than or equal to its associated single largest post-accident load and:

a) Following load rejection, the frequency is ≤ 63 Hz,

b) Within 2.2 seconds following load rejection, the voltage is ≥ 3740 V and ≤ 4580 V, and

c) Within 2.2 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz.

4.

> NOTE
>
> If performed with the diesel generator synchronized with offsite power, this surveillance shall be performed at a power factor ≤ 0.83 lagging. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.

Verifying the diesel generator does not trip following a load rejection of ≥ 2475 kW and ≤ 2750 kW.

SURVEILLANCE REQUIREMENTS (Continued)

5.

NOTE
This surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.

Verifying on an actual or simulated loss of offsite power in conjunction with an actual or simulated Engineered Safety Feature actuation signal:

a) De-energization of emergency buses,

b) Load shedding from emergency buses,

c) Diesel generator auto-starts from standby condition, and:

1. energizes permanently connected loads in ≤ 15 seconds,
2. energizes auto-connected loads through the load sequencer,
3. achieves steady state voltage ≥ 3740 V and ≤ 4580 V,
4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz and,
5. energizes permanently connected and auto-connected loads for ≥ 5 minutes.

6.

NOTE
This surveillance shall not normally be performed in MODE 1, 2, 3 or 4. However, this surveillance may be performed to reestablish OPERABILITY provided an assessment determines that safety of the plant is maintained or enhanced.

Verifying diesel generator automatic trips are bypassed on an actual or simulated loss of offsite power in conjunction with an actual or simulated Engineered Safety Feature actuation signal except:

a) Engine overspeed,

b) Generator differential current,

c) Voltage restraint overcurrent, and

d) Low lube oil pressure (switches to 2 out of 3 logic).

SURVEILLANCE REQUIREMENTS (Continued)

7.

NOTES

1. This surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.

2. The start of the diesel generator from a standby condition is not required if this surveillance is performed in conjunction with SR 4.8.1.1.2.c.5.

Verifying on an actual or simulated loss of offsite power signal:

a) De-energization of emergency buses,

b) Load shedding from emergency buses,

c) Diesel generator auto-starts from standby condition and:

1. energizes permanently connected loads in ≤ 15 seconds,

2. energizes auto-connected loads through the load sequencer,

3. achieves steady state voltage ≥ 3740 V and ≤ 4580 V,

4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz and,

5. energizes permanently connected and auto-connected loads for ≥ 5 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

8. Verifying on an actual or simulated Engineered Safety Feature actuation signal the diesel generator auto-starts from a standby condition and:

   a) Achieves ≥ 90% of rated speed and ≥ 97% of rated voltage in ≤ 15 seconds,

   b) Achieves steady state voltage ≥ 3740 V and ≤ 4580 V,

   c) Achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz,

   d) Operates for ≥ 5 minutes,

   e) Permanently connected loads remain energized from the offsite power system, and

   f) Auto-connected loads remain energized from the offsite power system as appropriate for plant conditions.

9.

> NOTE
>
> This surveillance shall be performed within 5 minutes of shutting down the diesel generator after the diesel generator has operated ≥1 hour loaded ≥ 2475 kW and ≤ 2750 kW. Momentary transients outside the load range do not invalidate this test.

Verifying the diesel generator starts and:

   a) Accelerates to ≥ 90% of rated speed and ≥ 97% of rated voltage in ≤ 15 seconds, and

   b) Achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.

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

1. Verifying the diesel starts from standby conditions and accelerates to ≥ 90% of rated speed and to ≥ 97% of rated voltage within 15 seconds after the start signal.

2. Verifying the generator achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.

3.

NOTES

1. Diesel generator loading may include gradual loading as recommended by the manufacturer.
2. Momentary transients outside the load range do not invalidate this test.
3. This test shall be conducted on only one diesel generator at a time.
4. This test shall be preceded by and immediately follow without shutdown a successful performance of SRs 4.8.1.1.2.d.1 and 4.8.1.1.2.d.2, or SR 4.8.1.1.2.a.2.

Verifying the diesel generator is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.

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 a fuel oil supply tank containing a minimum of 12,000 gallons of fuel.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could result in loss of required SDM or boron concentration, and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1 and 4.8.1.1.2, except for testing pursuant to Surveillance Requirements 4.8.1.1.2.a.3, 4.8.1.1.2.c.2, 4.8.1.1.2.c.5, 4.8.1.1.2.c.6, 4.8.1.1.2.c.7, and 4.8.1.1.2.d.3.

3/4 8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generators with tie breakers open between redundant busses:

4160	volt Emergency Bus # 24 C
4160	volt Emergency Bus #24 D
480	volt Emergency Load Center #22 E
480	volt Emergency Load Center #22 F
120	volt A.C. Vital Bus # VA-10
120	volt A.C. Vital Bus # VA-20
120	volt A.C. Vital Bus # VA-30
120	volt A.C. Vital Bus # VA-40

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus and/ or associated load center to OPERABLE status within 8 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from normal A.C. sources with tie breakers open between redundant busses at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

LIMITING CONDITION FOR OPERATION (Continued)

3.8.2.1A Inverters 5 and 6 shall be OPERABLE and available for automatic transfer via static switches VS1 and VS2 to power busses VA-10 and VA-20, respectively.

APPLICABILITY: MODES 1, 2 & 3

ACTION:

a. With inverter 5 or 6 inoperable, restore the inverter to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

b. With inverter 5 or 6 unavailable for automatic transfer via static switch VS1 or VS2 to power bus VA-10 or VA-20, respectively, restore the automatic transfer capability within 7 days or be in HOT SHUTDOWN within the next 12 hours.

c. With inverters 5 and 6 inoperable or unavailable for automatic transfer via static switches VS1 and VS2 to power busses VA-10 and VA-20, respectively, restore the inverters to OPERABLE status or restore their automatic transfer capability within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1A

a. Verify correct inverter voltage, frequency, and alignment for automatic transfer via static switches VS1 and VS2 to power busses VA-10 and VA-20, respectively, at the frequency specified in the Surveillance Frequency Control Program.

b. Verify that busses VA-10 and VA-20 automatically transfer to their alternate power sources, inverters 5 and 6, respectively, at the frequency specified in the Surveillance Frequency Control Program during shutdown.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator:

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Load Center

2 - 120 volt A.C. Vital Busses

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could result in loss of required SDM or boron concentration, and movement of recently irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE and energized from normal A.C. sources at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 125-volt D.C. bus Train A and 125-volt D.C. bus Train B electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one 125-volt D.C. bus train inoperable, restore the inoperable 125-volt D.C. bus train to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each 125-volt D.C. bus train shall be determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

4.8.2.3.2 Each 125-volt D.C. battery bank and charger of Train A and Train B shall be demonstrated OPERABLE:

a. By verifying at the frequency specified in the Surveillance Frequency Control Program that the battery cell parameters meet Table 4.8-1 Category A limits.

b. By verifying at the frequency specified in the Surveillance Frequency Control Program the battery cell parameters meet Table 4.8-1 Category B limits.

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 deterioration that could degrade battery performance,

   2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material, and

   3. The battery charger will supply at least 400 amperes at a minimum of 130 volts for at least 12 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 emergency loads for 8 hours 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.

SURVEILLANCE REQUIREMENTS (Continued)

Table 4.8-1
Battery Cell Parameters

Parameter	Category A: Limits For Each Designated Pilot Cell	Category B: Limits For Each Connected Cell
Electrolyte Level	Between the minimum and maximum level indication marks [a]	Not required.
Cell Voltage	$\geq$ 2.08 Volts	$\geq$ 2.08 Volts under float charge
Specific Gravity [b][c]	$\geq$1.200 (Corrected to 77°F)	$\geq$1.200 (Corrected to 77°F)
Battery Voltage	$\geq$ 125 Volts (Overall voltage)	Not required.

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during an equalizing charge provided it is not overflowing. Electrolyte level readings will be verified to meet the Category A limits within 7 days of completing an equalizing charge.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 5 amps when on float charge.

(c) A battery charging current of < 5 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 One 125 - volt D.C. bus train electrical power subsystem shall be OPERABLE:

APPLICABILITY: MODES 5 and 6.

ACTION:

With no 125-volt D.C. bus trains OPERABLE, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could result in loss of required SDM or boron concentration, and movement of recently irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus train shall be determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt D.C. bus train battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION SYSTEMS (TURBINE BATTERY) — OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.5 The Turbine Battery 125-volt D.C. electrical power subsystem shall be OPERABLE.

APPLICABILITY: MODES 1, 2 & 3

ACTION:

a. With the Turbine Battery 125-volt D.C. electrical power subsystem inoperable, restore the subsystem to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.5.1 Verify 125-volt D.C. bus 201D is OPERABLE at the frequency specified in the Surveillance Frequency Control Program.

4.8.2.5.2 125-volt D.C. battery bank 201D shall be demonstrated OPERABLE:

a. By verifying at the frequency specified in the Surveillance Frequency Control Program that the battery cell parameters meet Table 4.8-2 Category A limits.

b. By verifying at the frequency specified in the Surveillance Frequency Control Program the battery cell parameters meet Table 4.8-2 Category B limits.

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 deterioration that could degrade battery performance, and

   2. The cell-to-cell and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.

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 loads for 1 hour 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.

Table 4.8-2
Turbine Battery Cell Parameters

Parameter	Category A: Limits For Each Designated Pilot Cell	Category B: Limits For Each Connected Cell
Electrolyte Level	Between the minimum and maximum level indication marks [a]	Not required.
Cell Voltage	$\geq$ 2.08 Volts	$\geq$ 2.08 Volts under float charge
Specific Gravity [b][c]	$\geq$ 1.200 (Corrected to 77°F)	$\geq$ 1.200 (Corrected to 77°F)
Battery Voltage	$\geq$ 125 Volts (Overall Voltage)	Not required.

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during an equalizing charge provided it is not overflowing. Electrolyte level readings will be verified to meet the Category A limits within 7 days of completing an equalizing charge.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 5 amps when on float charge.

(c) A battery charging current of < 5 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATIONS

LIMITING CONDITION FOR OPERATION

3.9.1 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 following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 1720 ppm.

APPLICABILITY: MODE 6.

NOTE Only applicable to the refueling canal when connected to the Reactor Coolant System

ACTION:

With the requirements of the above specification not satisfied, within 15 minutes suspend all operations involving CORE ALTERATIONS and positive reactivity additions and initiate and continue boration at greater than or equal to 40 gpm of boric acid solution at or greater than the required refueling water storage tank concentration (ppm) 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 1720 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 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 CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of all filled portions of the reactor coolant system and the refueling canal shall be determined by chemical analysis at the frequency specified in the Surveillance Frequency Control Program.

4.9.1.3 Deleted

REFUELING OPERATIONS

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 Two source range neutron flux monitors shall be OPERABLE, each 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.

b. With both of the above required monitors inoperable, immediately initiate action to restore one monitor to OPERABLE status. Additionally, determine that the boron concentration of the Reactor Coolant System satisfies the requirements of LCO 3.9.1 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. Deleted

b. A CHANNEL CALIBRATION at the frequency specified in the Surveillance Frequency Control Program

c. A CHANNEL CHECK and verification of audible counts at the frequency specified in the Surveillance Frequency Control Program.

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3.1 The reactor shall be subcritical for a minimum of 100 hours prior to movement of irradiated fuel in the reactor pressure vessel.

APPLICABILITY: MODE 6.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3.1 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 pressure vessel.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

a. The equipment door 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. The personnel air lock shall be either:

1. closed by one personnel air lock door, or

2. capable of being closed by an OPERABLE personnel air lock 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 a manual or automatic isolation valve, blind flange, or equivalent, or

2. Be capable of being closed under administrative control *

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of irradiated fuel assemblies in the containment.

---

* Administrative controls shall ensure that appropriate personnel are aware that the equipment door, personnel air lock door and/or other containment penetrations are open, and that a specific individual(s) is designated and available to close the equipment door, personnel air lock 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 door, a personnel air lock door and/or other containment penetration must be capable of being quickly removed.

REFUELING OPERATIONS

CONTAINMENT PENETRATIONS

SURVEILLANCE REQUIREMENTS

4.9.4.1 Verify each required containment penetration is in the required status at the frequency specified in the Surveillance Frequency Control Program.

4.9.4.2 Deleted

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

SHUTDOWN COOLING AND COOLANT CIRCULATION - HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 One shutdown cooling train shall be OPERABLE and in operation.

NOTE

1. The required shutdown cooling train may not be in operation for up to 1 hour per 8 hour period provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1.

2. The normal or emergency power source may be inoperable for the required shutdown cooling train.

3. The shutdown cooling pumps may be removed from operation during the time required for local leak rate testing of containment penetration number 10 or to permit maintenance on valves located in the common SDC suction line, provided:

   a. No operations are permitted that would cause reduction of the Reactor Coolant System boron concentration,

   b. CORE ALTERATIONS are suspended, and

   c. Containment penetrations are in the following status:

      1) The equipment door is closed and secured with at least four bolts; and

      2) At least one personnel airlock door is closed; and

      3) Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed with a manual or automatic isolation valve, blind flange, or equivalent.

APPLICABILITY: MODE 6 with the water level $\geq$ 23 feet above the top of the reactor vessel flange.

SHUTDOWN COOLING AND COOLANT CIRCULATION - HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

ACTION:

With no shutdown cooling train OPERABLE or in operation, perform the following actions:

a. Immediately 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 and the loading of irradiated fuel assemblies in the core; and

b. Immediately initate action to restore one shutdown cooling train to OPERABLE status and operation; and

c. Within 4 hours place the containment penetrations in the following status:

    1. Close the equipment door and secure with at least four bolts; and

    2. Close at least one personnel airlock door; and

    3. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed with a manual or automatic isolation valve, blind flange, or equivalent.

SURVEILLANCE REQUIREMENTS

4.9.8.1.1 One shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate greater than or equal to 1000 gpm at the frequency specified in the Surveillance Frequency Control Program.

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

SHUTDOWN COOLING AND COOLANT CIRCULATION - LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two shutdown cooling trains shall be OPERABLE and one shutdown cooling train shall be in operation.

NOTE
The normal or emergency power source may be inoperable for each shutdown cooling train.

APPLICABILITY: MODE 6 with the water level < 23 feet above the top of the reactor vessel flange.

ACTION:

a. With one shutdown cooling train inoperable, immediately initiate action to restore the shutdown cooling train to OPERABLE status OR immediately initiate action to establish ≥ 23 feet of water above the top of the reactor vessel flange.

b. With no shutdown cooling train OPERABLE or in operation, perform the following actions:

1. Immediately 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; and

2. Immediately initiate action to restore one shutdown cooling train to OPERABLE status and operation; and

3. Within 4 hours place the containment penetrations in the following status:

    a. Closed the equipment door and secure with at least four bolts; and

    b. Close at least one personnel airlock door; and

LIMITING CONDITION FOR OPERATION (continued)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed with a manual or automatic isolation valve, blind flange, or equivalent.

SURVEILLANCE REQUIREMENTS

4.9.8.2.1 One shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate greater than or equal to 1000 gpm at the frequency specified in the Surveillance Frequency Control Program.

4.9.8.2.2 The required shutdown cooling 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 available.

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

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

REFUELING OPERATIONS

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 23.0 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts.

During movement of irradiated fuel assemblies within containment.

ACTION:

With the water level less than that specified above, immediately suspend CORE ALTERATIONS and immediately suspend movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level shall be determined to be within its minimum depth at the frequency specified in the Surveillance Frequency Control Program.

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.12 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: WHENEVER IRRADIATED FUEL ASSEMBLIES ARE IN THE STORAGE POOL.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel and spent fuel pool platform crane operations with loads in the fuel storage areas.

SURVEILLANCE REQUIREMENTS

4.9.12 The water level in the storage pool shall be determined to be within its minimum 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

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

REFUELING OPERATIONS

SHIELDED CASK

LIMITING CONDITION FOR OPERATION

3.9.16 All fuel within a distance L from the center of the spent fuel pool cask laydown area shall have decayed for at least 90 days. The distance L equals the major dimension of the shielded cask.

APPLICABILITY: Whenever a shielded cask is on the refueling floor.

ACTION:

With the requirements of the above specification not satisfied, do not move a shielded cask to the refueling floor. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.16 The decay time of all fuel within a distance L from the center of the spent fuel pool cask laydown area shall be determined to be ≥ 90 days within 24 hours prior to moving a shielded cask to the refueling floor and at the frequency specified in the Surveillance Frequency Control Program thereafter.

THIS PAGE INTENTIONALLY LEFT BLANK

SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.17 The boron concentration in the spent fuel pool shall be greater than or equal to 2100 parts per million (ppm).

APPLICABILITY: Whenever any fuel assembly or Non-standard Fuel Configuration is stored in the spent fuel pool.

ACTION:

With the boron concentration less than 2100 ppm, suspend the movement of all fuel assemblies, Non-standard Fuel Configurations, and shielded casks, and immediately initiate action to restore the spent fuel pool boron concentration to within its limit.

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.17 Verify that the boron concentration is greater than or equal to 2100 ppm at the frequency specified in the Surveillance Frequency Control Program, and within 24 hours prior to the initial movement of a fuel assembly or Non-standard Fuel Configuration in the Spent Fuel Pool, or shielded cask over the cask laydown area.

LIMITING CONDITION FOR OPERATION

3.9.18 The following spent fuel pool storage requirement will be met:

(a) Region 1 fuel assemblies have a maximum initial planar average enrichment of 4.85 weight percent of U-235. A fuel rod shall have a maximum enrichment of 5.0 weight percent of U-235. No burnup credit is required.

(b) Region 2 has two types of storage locations:

(1) The combination of initial planar average enrichment and burnup of a fuel assembly stored in Region 2 Type 2A shall be within the acceptable burnup domain of Figure 3.9-1A.

(2) The combination of initial planar average enrichment and burnup of a fuel assembly stored in Region 2 Type 2B shall be within the acceptable burnup domain of Figure 3.9-1B.

(c) Fuel assemblies stored in Region 3 shall contain either Borated Stainless Steel Poison Rodlets or a full length, full strength Control Element Assembly:

(1) The combination of initial planar average enrichment and burnup of a fuel assembly containing Borated Stainless Steel Poison Rodlets stored in Region 3 shall be within the acceptable burnup domain of Figure 3.9-1C. The Borated Stainless Steel Poison Rodlets shall be installed in the assembly's center guide tube and in two diagonally opposite guide tubes.

(2) The combination of initial planar average enrichment and burnup of a fuel assembly containing a full length, full strength Control Element Assembly stored in Region 3 shall be within the acceptable burnup domain of Figure 3.9-1D.*

(d) The combination of initial planar average enrichment and burnup of a fuel assembly stored in Region 4 shall be within the acceptable burnup domain of Figure 3.9-1E.

* Full-length, reduced-strength Control Element Assemblies and part-length Control Element Assemblies shall NOT be used in Region 3.

LIMITING CONDITION FOR OPERATION

(e) Each Non-standard Fuel Configuration must have a separate criticality analysis to determine where it can be stored in the spent fuel racks. The analysis may qualify storage in one or multiple Regions, and may or may not require Borated Stainless Steel Poison Rodlets or a full length, full strength Control Element Assembly if stored in Region 3.

APPLICABILITY: Whenever any fuel assembly or Non-standard Fuel Configuration is stored in the spent fuel pool.

ACTION:

Immediately initiate action to move the non-complying fuel assembly or Non-standard Fuel Configuration to an acceptable location.

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.18 Prior to storing a fuel assembly in the spent fuel racks, verify by administrative means the initial planar average enrichment and burnup of the fuel assembly is in accordance with the acceptable specifications for that Storage Region. Prior to storing a Non-standard Fuel Configuration in the spent fuel racks, verify by administrative means the Non-standard Fuel Configuration is qualified for that Storage Region.

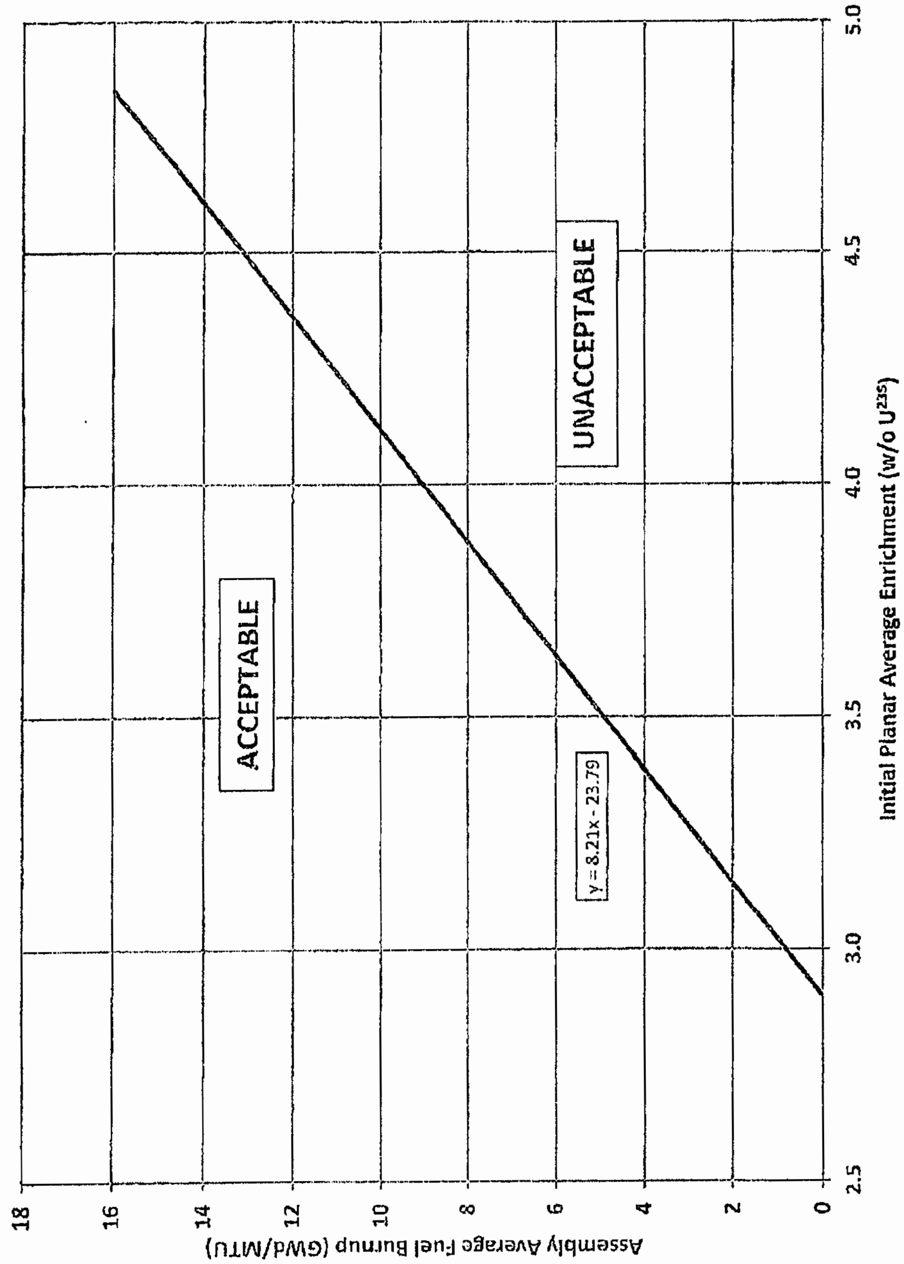


FIGURE 3.9-1A MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2A

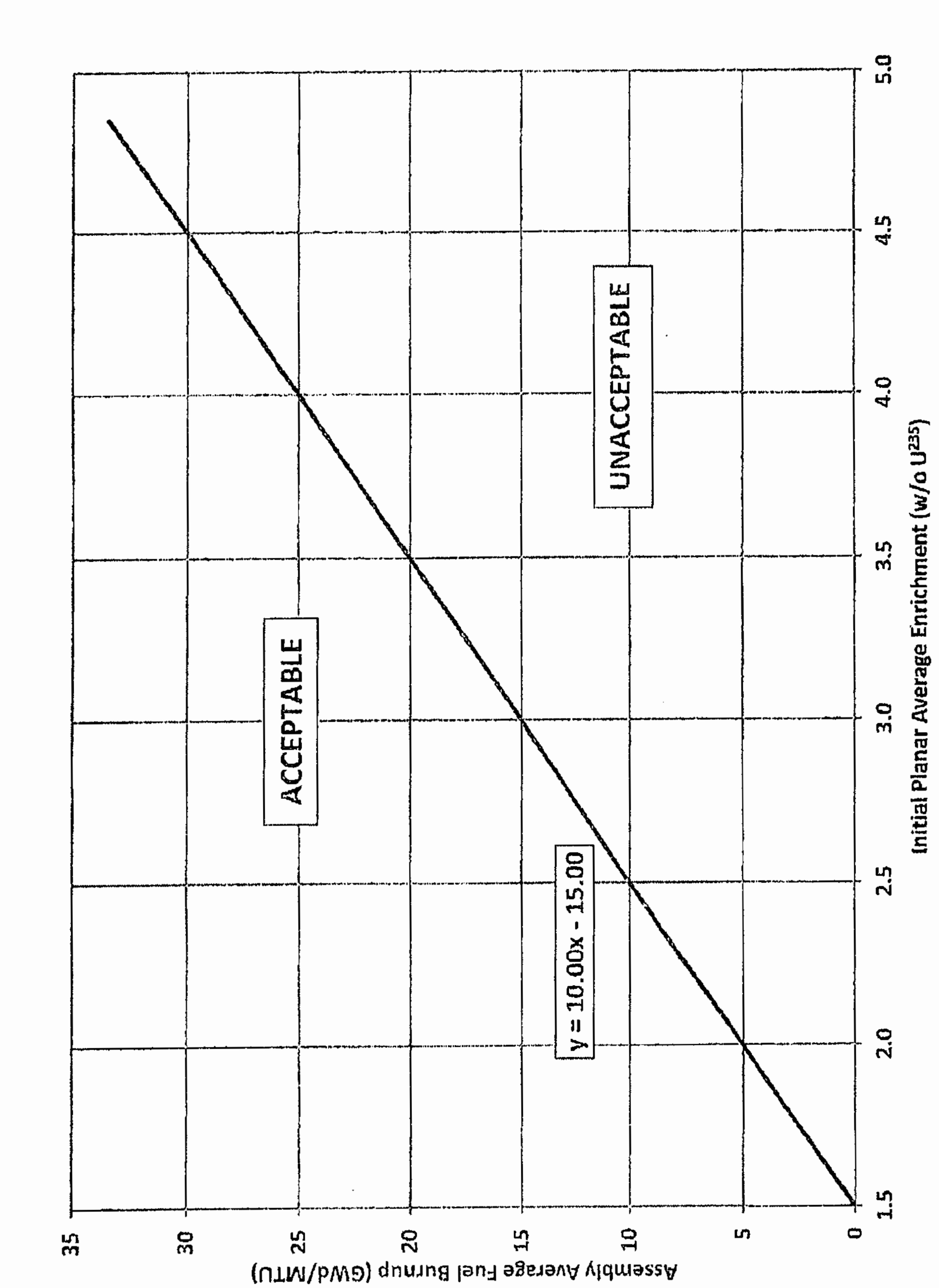


FIGURE 3.9-1B MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2B

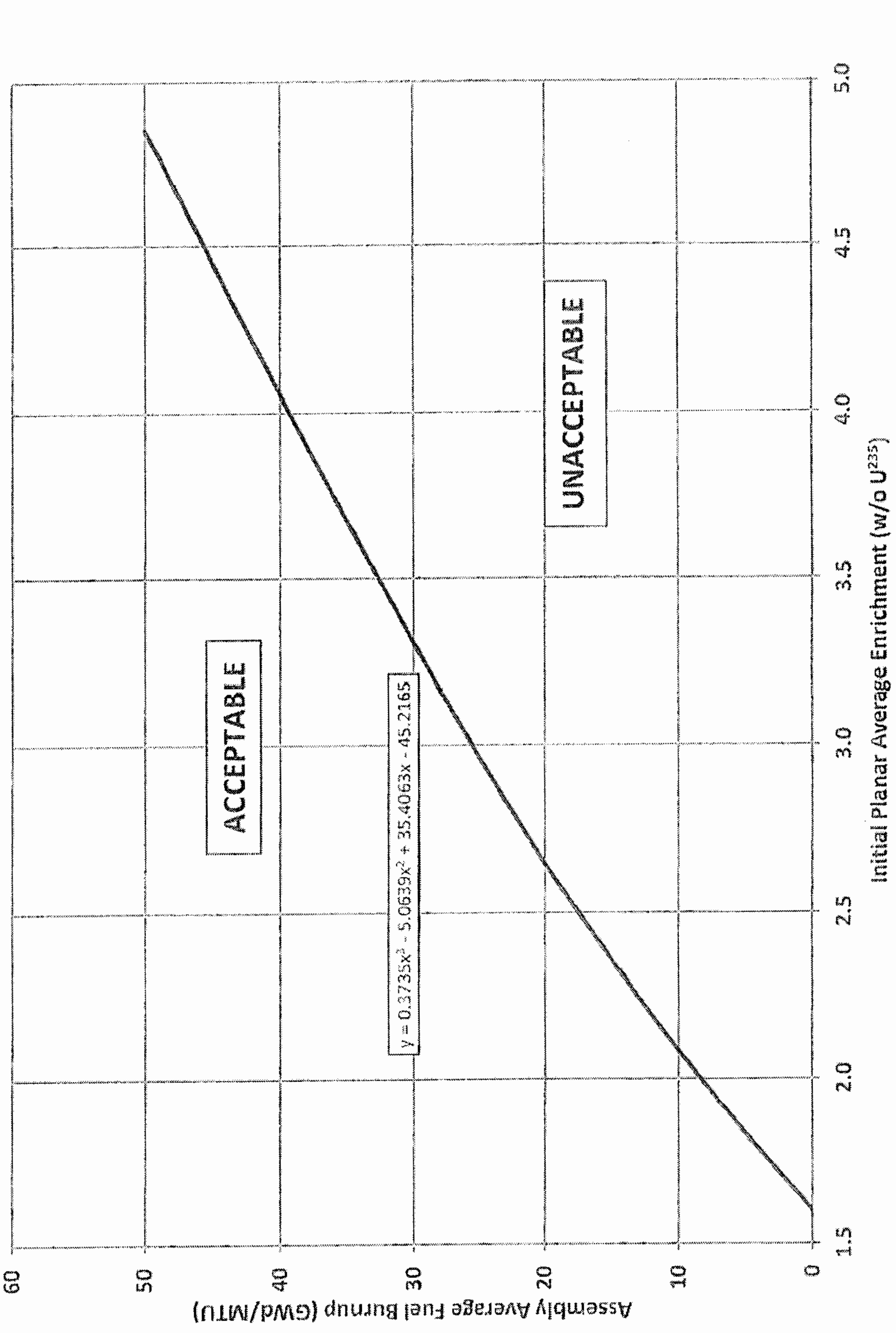


Figure 3.9-1C MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 3
(with insertion of 3 Borated Stainless Steel Poison Rodlets)

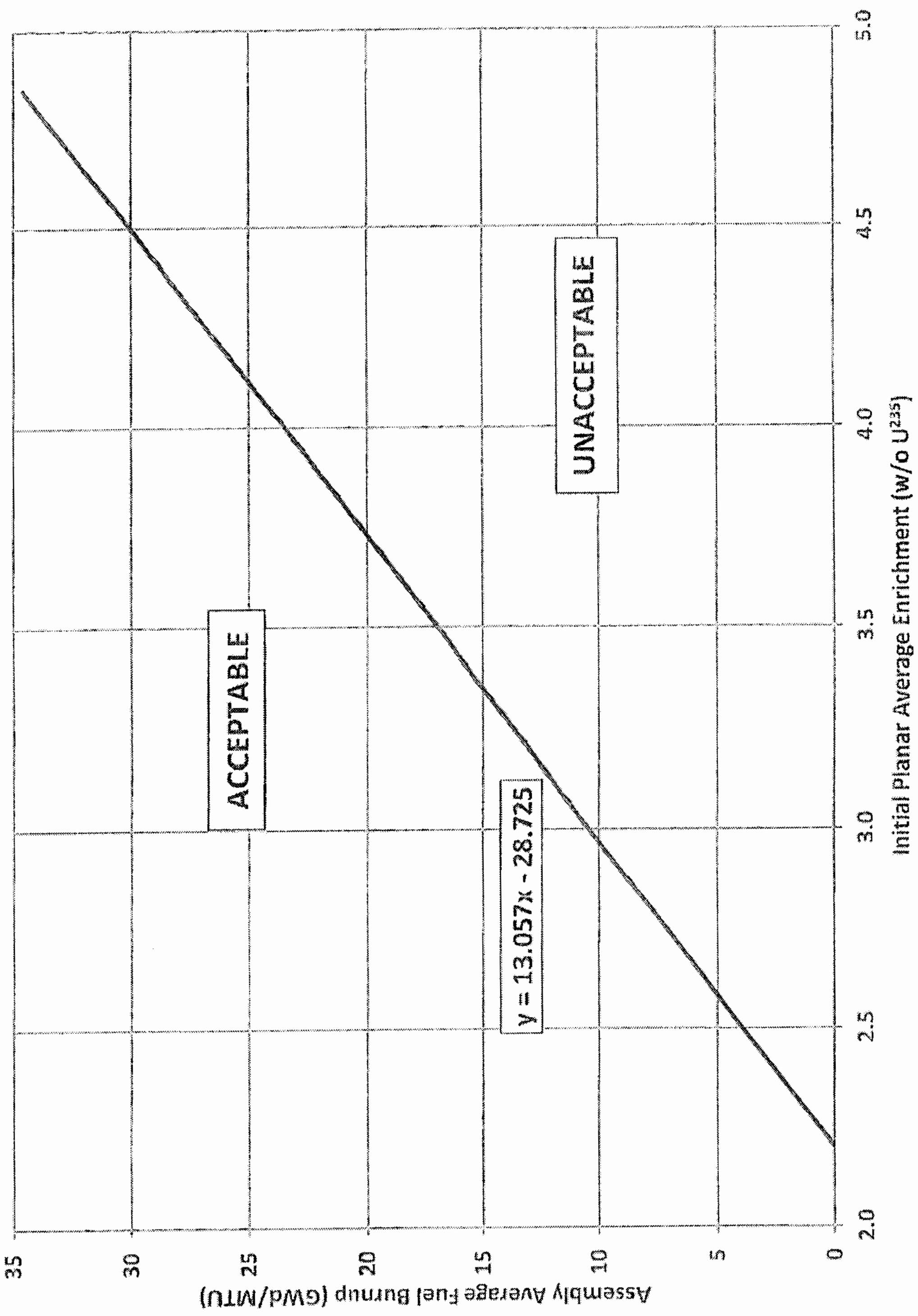


Figure 3.9-1D MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 3 (with insertion of a full length, full strength Control Element Assembly)

Revised by NRC correction letter dated September 19, 2016

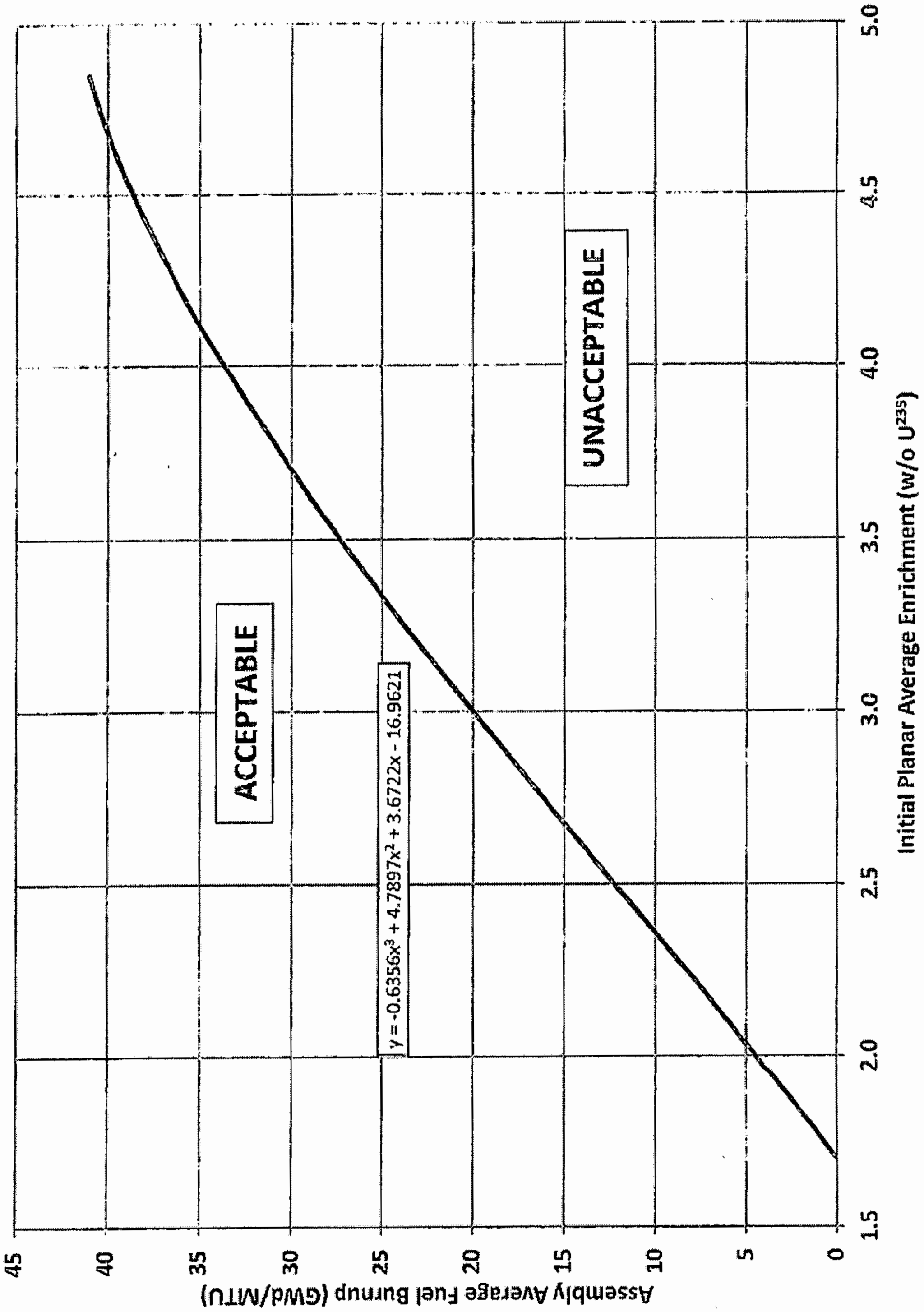
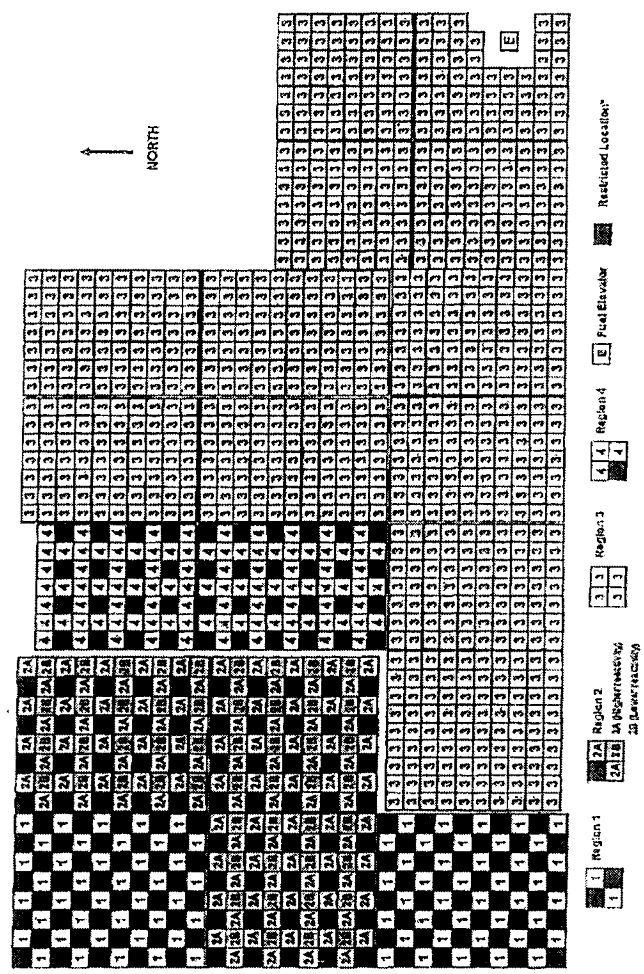


FIGURE 3.9-1E MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 4

SPENT FUEL POOL ARRANGEMENT
FIGURE 3.9-2
(NOT TO SCALE)



* A Restricted Location shall remain empty. No fuel assembly, no Non-standard Fuel Configuration, no non-fuel component, nor any hardware/material of any kind may be stored in a Restricted Location.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

SPENT FUEL POOL - STORAGE RESTRICTIONS

LIMITING CONDITION FOR OPERATION

3.9.19 The following spent fuel pool storage restrictions will be met:

(1) Restricted Locations shall remain empty. No fuel assembly, no Non-standard Fuel Configuration, no non-fuel component, nor any hardware/material of any kind may be stored in a Restricted Location (shown in Figure 3.9-2).

(2) Fuel assemblies and Non-standard Fuel Configurations shall NOT be stored in Region 1 and 2 storage locations in which the Boraflex panel box has been removed. It is permissible to store non-fuel components in non-restricted locations with or without a Boraflex panel box. *

APPLICABILITY: Fuel assemblies, Non-standard Fuel Configurations, or non-fuel components in the spent fuel pool.

ACTION:

Take immediate action to comply with either 3.9.19(1) or (2).

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.19 Verify that 3.9.19 is satisfied prior to storing fuel assemblies, Non-standard Fuel Configurations, or non-fuel components in the spent fuel racks.

* Note that Region 1 and 2 spent fuel pool rack storage locations contain removable Boraflex panel boxes which house the Boraflex panels. The Boraflex panel boxes were manufactured as an integral part of the original spent fuel pool racks and as such are NOT stored components in SFP rack storage locations. Criticality analysis has shown that the Restricted Locations are acceptable with or without the Boraflex panel boxes.

THIS PAGE INTENTIONALLY LEFT BLANK

REFUELING OPERATIONS

SPENT FUEL POOL - CONSOLIDATION

LIMITING CONDITION FOR OPERATION

3.9.20 Prior to consolidation of spent fuel assemblies, the candidate fuel assemblies must have decayed for at least 5 years.

APPLICABILITY: During all consolidation operations.

ACTION:

With the requirements of the above specification not satisfied, replace candidate assembly with an appropriate substitute or suspend all consolidation activities.

SURVEILLANCE REQUIREMENTS

4.9.20 The decay time of all candidate fuel assemblies for consolidation shall be determined to be greater than or equal to five years within 7 days prior to moving the fuel assembly into the consolidation work station.

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The requirement of Specifications 3.1.1.1, 3.1.3.5 and 3.1.3.6 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3[1] during PHYSICS TESTS.

ACTION:

a. With any CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, within 15 minutes initiate and continue boration at > 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at > 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position once within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1[2].

---

[1] Operation in MODE 3 shall be limited to 6 consecutive hours.

[2] Not required to be performed during initial power escalation following a refueling outage if SR 4.1.3.4 has been met.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT AND INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 are suspended, immediately:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1 or

b. Be in HOT STANDBY within 2 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at the frequency specified in the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 or 3.2.4 are suspended.

"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

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE LOCATION

The Unit 2 Containment Building is located on the site at Millstone Point in Waterford, Connecticut. The nearest SITE BOUNDARY on land is 2034 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

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing 176 rods. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum initial planar average enrichment of 4.85 weight percent of U-235. A fuel rod shall have a maximum enrichment of 5.0 weight percent of U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 control element assemblies. The control element assemblies shall be designed and maintained in accordance with the design provisions contained in Section 3.0 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.4 DELETED

5.5 DELETED

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a) The new fuel (dry) storage racks are designed and shall be maintained with sufficient center to center distance between assemblies to ensure a $K_{eff} \leq .95$. The maximum initial planar average fuel assembly enrichment to be stored in these racks is 4.85 weight percent U-235. The maximum initial fuel rod enrichment to be stored in these racks is 5.0 weight percent of U-235.

b) The spent fuel storage racks are designed and shall be maintained with fuel assemblies having a maximum initial planar average enrichment of 4.85 weight percent U-235. The maximum initial fuel rod enrichment to be stored in these racks is 5.0 weight percent U-235.

c) The spent fuel storage racks are designed and shall be maintained with $K_{eff} < 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties and biases.

d) The spent fuel storage racks are designed and shall be maintained with $K_{eff} \leq .95$ if fully flooded with water borated to 600 ppm, which includes an allowance for uncertainties and biases.

e) Region 1 of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Region 1 contains the Restricted Locations, shown in Figure 3.9-2. Fuel having an initial planar average enrichment of 4.85 weight percent U-235 may be stored in available locations.

f) Region 2 of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Region 2 contains Type 2A and Type 2B storage locations as well as the Restricted Locations shown in Figure 3.9-2. Fuel assemblies stored in this region must comply with Figure 3.9-1A or Figure 3.9-1B. Fuel assemblies utilizing Figure 3.9-1A must be stored in the Region 2 Type 2A storage locations, and fuel assemblies utilizing Figure 3.9-1B must be stored in the Region 2 Type 2B storage locations.

g) Region 3 of the spent fuel storage pool is designed and shall be maintained with a nominal 9.0 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figure 3.9-1C or Figure 3.9-1D. Additionally, fuel assemblies utilizing Figure 3.9-1C require that Borated Stainless Steel Poison Rodlets be inserted in the fuel assembly's center guide tube and in two diagonally opposite guide tubes. The poison rods are solid nominal 0.87 inch O.D. borated stainless steel, with a nominal boron content of 2.0 weight percent boron. Finally, fuel assemblies utilizing Figure 3.9-1D require that a full length, full strength Control Element Assembly be inserted in the fuel assembly (full-length, reduced-strength Control Element Assemblies and part-length Control Element Assemblies shall NOT be used in Region 3).

h) Region 4 of the spent fuel storage pool is designed and shall be maintained with a nominal 9.0 inch center to center distance between storage locations. Region 4 contains Restricted Locations as shown in Figure 3.9-2. Fuel assemblies stored in this region must comply with Figure 3.9-1E.

i) Each region of the spent fuel storage pool is designed to permit storage of Non-standard Fuel Configurations, except for the Restricted Locations. Each of the Non-standard Fuel Configurations must have a separate criticality analysis which may allow storage in one or multiple Regions, and which may or may not require Borated Stainless Steel Poison Rodlets or a full length, full strength Control Element Assembly if stored in Region 3.

j) Regions 1 and 2 spent fuel racks are equipped with boxes that contain the Boraflex panels which may be removed from both non-restricted and Restricted Locations. Fuel assemblies and Non-standard Fuel Configurations shall NOT be placed in storage locations in which the Boraflex panel box has been removed (however, it is permissible to store non-fuel components in a location in which the Boraflex panel box has been removed as long as the location is NOT a Restricted Location).

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 22'6".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with the number of storage locations (including Restricted Locations) limited to no more than 160 storage locations in Region 1, 224 storage locations in Region 2, 822 storage locations in Region 3, and 140 storage locations in Region 4 for a total of 1346 storage locations.

5.7 DELETED

5.8 DELETED

5.9 DELETED

50-336

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 higher 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 operation, 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.

6.2.2 FACILITY STAFF (Continued)

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

FACILITY STAFF (CONTINUED)

d. A radiation protection technician shall be on site when fuel is in the reactor. (Table 6.2-1)

e. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. Deleted

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. 2.

THIS PAGE INTENTIONALLY LEFT BLANK

TABLE 6.2-1[3]

MINIMUM SHIFT-CREW COMPOSITION[2]

LICENSE CATEGORY	APPLICABLE MODES 1, 2, 3 & 4	APPLICABLE MODES 5 & 6
Senior Reactor Operator	2	1[1]
Reactor Operator	2	1
Non-Licensed Operator	2	1
Shift Technical Advisor	1[4]	None Required

(1) Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling individual supervising CORE ALTERATIONS after the initial fuel loading.

(2) The above shift crew composition and the qualified radiation protection technician of Section 6.2.2 may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided expeditious actions are taken to fill the required position.

(3) Requirements for minimum number of licensed operators on shift during operation in MODES other than COLD SHUTDOWN or REFUELING are contained in 10CFR50.54(m).

(4) The Shift Technical Advisor position can be filled by either of the two Senior Reactor Operators (a dual-role individual), if he meets the Shift Technical Advisor qualifications of the Commission Policy Statement on Engineering Expertise on Shift.

6.4 TRAINING

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.

6.5 Deleted.

PAGES 6-6 THROUGH 6-13 HAVE BEEN INTENTIONALLY DELETED.

6.6 Deleted.

6.7 Deleted.

6.8 PROCEDURES

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, February, 1978.

b. Refueling operations.

c. Surveillance activities of safety related equipment..

d. Not used.

e. Not used.

f. Fire Protection Program implementation.

g. Quality Control for effluent monitoring using the guidance in Regulatory Guide 1.21 Rev. 1, June 1974.

h. 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 the 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 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved in accordance with the Quality Assurance Program Topical Report within 14 days of implementation.

6.8.4 Written procedures shall be established, implemented and maintained covering Section I.E, Radiological Environmental Monitoring, of the REMODCM.

6.8.5 All procedures and procedure changes required for the Radiological Environmental Monitoring Program (REMP) of 6.8.4 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 plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR 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 additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1]

6.9.1.4 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

6.9.1.5a. DELETED

6.9.1.5b DELETED

6.9.1.5c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit. The report covering the previous calendar year shall be submitted prior to March 1 of each year.

1 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 RADIOLOGICAL REPORTS

6.9.1.6a 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.

6.9.1.6b 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.I.

6.9.1.7 Deleted

CORE OPERATING LIMITS REPORT

6.9.1.8 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.

3/4.1.1.1	SHUTDOWN MARGIN (SDM)
3/4.1.1.4	Moderator Temperature Coefficient
3/4.1.3.6	Regulating CEA Insertion Limits
3/4.2.1	Linear Heat Rate
3/4.2.3	TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r
3/4.2.6	DNB Margin

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1) EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 - Methodology Description, Volume 2 -Benchmarking Results," Siemens Power Corporation.

2) ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels.

3) XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company.

4) XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company.

5) EMF-2328(P)(A) and Supplement 1, "PWR Small Break LOCA Evaluation Model S-RELAP5 Based."

6) EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation.

7) XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company.

CORE OPERATING LIMITS REPORT (CONT.)

8) XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company.

9) XN-NF-82-06(P)(A), and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company.

10) ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.

11) XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company.

12) ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation.

13) EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation.

14) EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP.

15) EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation.

16) EMF-92-116(P)(A) and Supplement 1, "Generic Mechanical Design Criteria for PWR Fuel Designs."

17) BAW-10240(P)(A) Revision 0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," May 2004.

18) EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."

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.

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

6.9.1.9 A report shall be submitted within 180 days after initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.26, 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.

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

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. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Deleted

b. Deleted

c. Deleted

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Deleted

f. Deleted

g. RCS Overpressure Mitigation, Specification 3.4.9.3.

6.9.2 (Continued)

h. Deleted

i. Tendon Surveillance Report, Specification 6.25

j. Deleted

k. Accident Monitoring Instrumentation, Specification 3.3.3.8.

l. Radiation Monitoring Instrumentation, Specification 3.3.3.1.

m. Deleted

6.10 Deleted.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of 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.

6.12 HIGH RADIATION AREA (CONT.)

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 from entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.12 HIGH RADIATION AREA (CONT.)

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 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

6.12 HIGH RADIATION AREA (CONT.)

(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 SYSTEMS INTEGRITY

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

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

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

6.14 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

6.15 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 monitoring 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 6.9.1.6a and specification 6.9.1.6b.

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 of 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 indicate the date (i.e., month and year) the change was implemented.

6.16 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 utilizing the guidance provided in the REMODCM.

6.17 SECONDARY WATER CHEMISTRY

A program shall be maintained 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.

6.18 Deleted

*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.

6.19 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the primary containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in 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 primary Containment internal pressure for the design basis loss of coolant accident is P_a. P_a is 53 psig. Containment leakage rate testing will be performed at the containment design pressure of 54 psig or higher.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $< 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $< 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$ 25 psig.

The provisions of SR 4.0.2 do not apply for test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

6.20 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 10CFR 20.1001-20.2402;

6.20 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

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.

6.21 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provided (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 that 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.

6.22 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.

6.23 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 in the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.23.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).

6.24 DIESEL FUEL OIL TEST PROGRAM

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An American Petroleum Institute (API) gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. A clear and bright appearance with proper color or a water and sediment content within limits.

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/L when tested every 92 days.

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Diesel Fuel Oil Test Program test frequencies.

6.25 PRE-STRESSED CONCRETE CONTAINMENT TENDON SURVEILLANCE PROGRAM

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

The provisions of Surveillance Requirement 4.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

6.25 PRE-STRESSED CONCRETE CONTAINMENT TENDON SURVEILLANCE PROGRAM (CONTINUED)

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken. This Tendon Surveillance Report is an administrative requirement listed in Technical Specification 6.9.2, “Special Reports.”

6.26 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 combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

    2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gpd per SG.

    3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational LEAKAGE."

6.26 STEAM GENERATOR (SG) PROGRAM (Continued)

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.

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. 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 72 effective full power months or at least every third 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, c, and d 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 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 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.

6.26 STEAM GENERATOR (SG) PROGRAM (Continued)

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall 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 associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

6.27 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 E-741 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. The exceptions to the Regulatory Guides (RG) referenced in RG 1.196 (i.e., RG 1.52, RG 1.78, and 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. Licensee controlled programs will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from those programs will be entered into the corrective action process and shall be trended and used as part of the 36 month assessment of the CRE boundary in accordance with 6.27.c.(ii).

6.27 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM (Continued)

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of Surveillance Requirement 4.0.2 are applicable to the frequencies for assessing CRE habitability and determining CRE unfiltered inleakage as required by paragraph c.

6.28 SNUBBER EXAMINATION, TESTING, AND SERVICE LIFE MONITORING PROGRAM

This program conforms to the examination, testing, and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

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.

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.

6.29 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.

6.29 SURVEILLANCE FREQUENCY CONTROL PROGRAM (Continued)

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.

APPENDIX B
ADDITIONAL CONDITIONS
FACILITY OPERATING LICENSE NO. DPR-65

The licensee shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
212	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in Attachment 3 of the licensee's application dated September 3, 1997, and evaluated in the staff's Safety Evaluation dated January 23, 1998.	30 days from the date of issuance
222	This amendment authorizes the licensee to include in the Updated Final Safety Analysis Report (UFSAR) changes to the description of the facility. Implementation of this amendment is the updating of the UFSAR to reflect the changes in Attachment 3 of the licensee's application dated July 2, 1998, and evaluated in the staff's Safety Evaluation dated December 18,1998	Next UFSAR update