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

NEXTERA ENERGY SEABROOK, LLC, ET AL.*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE

Renewed License No. NPF-86

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

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

B. Construction of the Seabrook Station, Unit No. 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-135 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

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

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

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

F. The licensees have satisfied the applicable provisions of 10 CFR 140, “Financial Protection Requirements and Indemnity Agreements,” of the Commission's regulations;

* NextEra Energy Seabrook, LLC, is authorized to act as agent for the: Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, and Taunton Municipal Lighting Plant (collectively, with NextEra Energy Seabrook, LLC, “licensees”) and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

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

H. After weighing the environmental, economical, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Renewed Facility Operating License No. NPF-86, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR 51 of the Commission's regulations and all applicable requirements have been satisfied;

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

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

2. Based on the foregoing findings, Facility Operating License No. NPF-86, issued on March 15, 1990, is superseded by Renewed Facility Operating License No. NPF-86, which is hereby issued as follows:

A. This renewed license applies to the Seabrook Station, Unit No. 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in Seabrook Township, Rockingham County, on the southeast coast of the State of New Hampshire, and is described in the licensees' "Final Safety Analysis Report," as supplemented and amended, and in the licensees' Environmental Report, as supplemented and amended.

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

(1) NextEra Energy Seabrook, LLC, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in Rockingham County, New Hampshire, in accordance with the procedures and limitations set forth in this renewed license;

(2) Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, and Taunton Municipal Lighting Plant to possess the facility at the designated location in Rockingham County, New Hampshire, in accordance with the procedures and limitations set forth in this renewed license;

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

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

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

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

(7) DELETED

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

(1) <u>Maximum Power Level</u>

NextEra Energy Seabrook, LLC, is authorized to operate the facility at reactor core power levels not in excess of 3648 megawatts thermal (100% of rated power).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 175, are incorporated into the Renewed Facility Operating License No. NPF-86. NextEra Energy Seabrook, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) License Transfer to FPL Energy Seabrook, LLC**

a. On the closing date(s) of the transfer of any ownership interests in Seabrook Station covered by the Order approving the transfer, FPL Energy Seabrook, LLC**, shall obtain from each respective transferring owner all of the accumulated decommissioning trust funds for the facility, and ensure the deposit of such funds and additional funds, if necessary, into a decommissioning trust or trusts for Seabrook Station established by FPL Energy Seabrook, LLC**, such that the amount of such funds deposited meets or exceeds the amount required under 10 CFR 50.75 with respect to the interest in Seabrook Station FPL Energy Seabrook, LLC**, acquires on such dates(s).

b. With respect to the decommissioning trust(s) established by FPL Energy Seabrook, LLC**,

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

(ii) Investments in the securities or other obligations of FPL Group Inc. or its affiliates, successors, or assigns shall be prohibited. In addition, except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants shall be prohibited.

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

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

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

** On April 16, 2009, the name “FPL Energy Seabrook, LLC” was changed to “NextEra Energy Seabrook, LLC.”

c. NextEra Energy Seabrook, LLC, shall take all necessary steps to ensure that the decommissioning trust(s) are maintained in accordance with the license transfer application and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order.

d. NextEra Energy Seabrook, LLC, shall take no action to cause FPL Group Capital, Inc. or its parent companies to void, cancel, or modify the Support Agreement to provide funding of up to $110 million for FPL Energy Seabrook, LLC**, as represented in the license transfer application without prior written consent of the Director of the Office of Nuclear Reactor Regulation.

(4) 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:

(i) Pre-defined coordinated fire response strategy and guidance
(ii) Assessment of mutual aid fire fighting assets
(iii) Designated staging areas for equipment and materials
(iv) Command and control
(v) Training of response personnel

b. Operations to mitigate fuel damage considering the following:

(i) Protection and use of personnel assets
(ii) Communications
(iii) Minimizing fire spread
(iv) Procedures for implementing integrated fire response strategy
(v) Identification of readily-available, pre-staged equipment
(vi) Training on integrated fire response strategy

c. Actions to minimize release to include consideration of:

(i) Water spray scrubbing
(ii) Dose to onsite responders

(5) License Renewal License Conditions

a. The information in the Final Safety Analysis Report (FSAR) supplement, submitted pursuant to 10 CFR 54.21(d) and as revised during the license renewal application review process, and licensee commitments as listed in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Seabrook Station," are collectively the "License Renewal FSAR Supplement." This Supplement is henceforth part of the FSAR,

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which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs, activities, and commitments described in this Supplement, without prior Commission approval, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, tests, and experiments," and otherwise complies with the requirements in that section.

b. The License Renewal FSAR Supplement, as defined in license condition (5)a above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).

(i) NextEra Energy Seabrook, LLC shall implement those new programs and enhancements to existing programs no later than 6 months before the PEO (September 15, 2029)

(ii) NextEra Energy Seabrook, LLC shall complete those activities by the date 6 months before the PEO (September 15, 2029) or by the end of the last refueling outage before the PEO, whichever occurs later.

(iii) NextEra Energy Seabrook, LLC shall notify the NRC in writing within 30 days after having accomplished item b(i) above and include the status of those activities that have been or remain to be completed in item b(ii) above.

D. Exemptions

NextEra Energy Seabrook, LLC is exempted from the Section III.D.2(b)(ii) containment airlock testing requirements of Appendix J to 10 CFR Part 50, because of the special circumstances described in Section 6.2.6 of SER Supplement 5 to the original operating license (NUREG-0896, dated July 1986) and authorized by 10 CFR 50.12(a)(2)(ii) and (iii) (51 FR 37684 October 23, 1986).

NRC Materials License No. SNM-1963, issued December 19, 1985, granted an exemption pursuant to 10 CFR 70.24 with respect to requirements for criticality alarms. NextEra Energy Seabrook, LLC, is hereby exempted from provisions of 10 CFR 70.24 insofar as this section applies to the storage and handling of new fuel assemblies in the new fuel storage vault, spent fuel pool (when dry), and shipping containers.

These exemptions, authorized by law, will not present an undue risk to the public health and safety and are consistent with the common defense and security. These exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Physical Security

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 provision 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[1], submitted by letter dated September 23, 2004, and supplemented by letters dated October 15, October 22, and October 29, 2004, and May 18, 2006, is entitled: "Florida Power and Light & FPL Energy Seabrook Physical Security Plan, Training and Qualification Plan and Safeguards Contingency Plan." The set contains Safeguards Information protected under 10 CFR 73.21. NextEra Energy Seabrook, LLC shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NextEra Energy Seabrook, LLC CSP was approved by License Amendment No. 127 (as supplemented by clarifications approved by License Amendment No. 132 and License Amendment No. 146).

F. Fire Protection

NextEra Energy Seabrook, LLC, shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report, the Fire Protection Program Report, and the Fire Protection of Safe Shutdown Capability report for the facility, as supplemented and amended, and as approved in the Safety Evaluation Report, dated March 1983; Supplement 4, dated May 1986; Supplement 5, dated July 1986; Supplement 6, dated October 1986; Supplement 7, dated October 1987; and Supplement 8, dated May 1989 subject to the following provisions: NextEra Energy Seabrook, LLC, may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain shutdown in the event of a fire.

G. Fixed Incore Detector Analysis

If the methodology described in Appendix B of ANP-3243P, Revision 1, "Seabrook Station, Unit 1 Fixed Incore Detector System Analysis Supplement to YAEC-1855PA," is utilized in any plant surveillance then NextEra must notify the NRC by letter of the plant's conditions and results of that surveillance.

H. Financial Protection

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

I. DELETED

J. Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 159, are hereby incorporated into this renewed license. NextEra Energy

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

Seabrook, LLC, shall operate the facility in accordance with the Additional Conditions.

K. Inadvertent Actuation of the Emergency Core Cooling System (ECCS)

Prior to startup from refueling outage 11, FPL Energy Seabrook** commits to either upgrade the controls for the pressurizer power operated relief valves (PORV) to safety-grade status and confirm the safety-grade status and water-qualified capability of the PORVs, PORV block valves and associated piping or to provide a reanalysis of the inadvertent safety injection event, using NRC approved methodologies, that concludes that the pressurizer does not become water solid within the minimum allowable time for operators to terminate the event. NextEra Energy Seabrook, LLC submitted an analysis of the inadvertent safety injection event in a letter dated November 7, 2005. In a letter dated June 9, 2006, the NRC concluded the analysis met the requirements of License Condition 2.K.

3. This renewed license is effective as of the date of issuance and shall expire at midnight on March 15, 2050.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ho K. Nieh, Director
Office of Nuclear Reactor Regulation

Attachments/ Appendices:
1. Appendix A - Technical Specifications (NUREG-1386)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional Conditions

Date of Issuance: March 12, 2019

** On April 16, 2009, the name "FPL Energy Seabrook, LLC" was changed to "NextEra Energy Seabrook, LLC."

NUREG-1386

Technical Specifications Seabrook Station, Unit 1

Docket No. 50-443

Appendix "A" to
License No. NPF-86

Issued by the
U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1990

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NUREG-1386

Technical Specifications
Seabrook Station,
Unit 1

Docket No. 50-443

Appendix "A" to
License No. NPF-86

Issued by the
U.S. Nuclear Regulatory Commission

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March 1990

INDEX

1.0 DEFINITIONS

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

ACTION

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

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy. The ANALOG CHANNEL OPERATIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

AXIAL FLUX DIFFERENCE

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

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

CHANNEL CHECK

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

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions.

b. All equipment hatches are closed and sealed,
c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
d. The containment leakage rates are in accordance with the Containment Leakage Rate Testing Program, and
e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or 0-rings) is OPERABLE.

CONTROLLED LEAKAGE

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

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement of any fuel, sources, 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.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) provides core operating limits for the current operating reload cycle. The cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.8.1.6. Plant operation within these operating limits is addressed in individual specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1.11 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation and/or injecting simulated process data to verify OPERABILITY of alarm and/or trip functions. The Digital Channel Operational Test definition is only applicable to the Radiation Monitoring Equipment. The DIGITAL CHANNEL OPERATIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same TEDE dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed under inhalation in Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample with half-lives greater than 10 minutes.

ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME

1.14 The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.16 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.17 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 either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

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

INSERVICE TESTING PROGRAM

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

MASTER RELAY TEST

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

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL

1.20 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.7.6 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.8.1.3 and 6.8.1.4.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.22 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.23 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM

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

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3648 Mwt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

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

REPORTABLE EVENT

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

1.31 (NOT USED)

SHUTDOWN MARGIN

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

SITE BOUNDARY

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

SLAVE RELAY TEST

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

1.35 (NOT USED)

SOURCE CHECK

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

STAGGERED TEST BASIS

1.37 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.39 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy. The TRIP ACTUATING DEVICE OPERATIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

1.42 (NOT USED)

VENTING

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

TABLE 1.1

FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL MODES

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

*Excluding decay heat.

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS (SLs)

2.1.1 REACTOR CORE SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

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

2.1.1.2 The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained less than or equal to 2735 psig.

2.1.3 SAFETY LIMIT VIOLATIONS

2.1.3.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.1.3.2 If SL 2.1.2 is violated:

a. In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

b. In MODE 3, 4, or 5, restore compliance within 5 minutes.

Figure 2.1-1 (THIS FIGURE IS NOT USED)

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

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

ACTION:

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

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

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

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

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

   Where:

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

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

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

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

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	1.42	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	1.42	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. (NOT USED)					
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤31.1% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.6 \times 10^5$ cps
7. Overtemperature ΔT	N.A.	N.A.	N.A.	See Note 1	See Note 2
8. Overpower ΔT	N.A.	N.A.	N.A.	See Note 3	See Note 4
9. Pressurizer Pressure - Low	N.A.	N.A.	N.A.	≥1945 psig	≥1,933 psig, See Note 5
10. Pressurizer Pressure - High	N.A.	N.A.	N.A.	≤2385 psig	≤2,397 psig

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. Pressurizer Water Level - High	8.0	4.20	0.84	≤92% of instrument span	≤93.75% of instrument span
12. Reactor Coolant Flow - Low	N.A.	N.A.	N.A.	≥90% of indicated loop flow	≥89.6% of indicated loop flow
13. Steam Generator Water Level Low - Low	N.A.	N.A.	N.A.	≥20.0% of narrow range instrument span	≥19.5% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	15.0	1.39	0	≥10,200 volts	≥9,822 volts
15. Underfrequency - Reactor Coolant Pumps	2.9	0	0	≥55.5 Hz	≥55.3 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥500 psig	≥450 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤10% of RTP*	≤12.1% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	≤10% RTP* Turbine Impulse Pressure Equivalent	≤12.3% of RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤50% of RTP*	≤52.1% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤45% of RTP*	≤45.3% of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	≥10% of RTP*	≥7.9% of RTP*
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	≤10% RTP* Turbine Impulse Pressure Equivalent	≤12.3% RTP* Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \frac{(1)}{(1 + \tau_6 S)} - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured RCS ΔT by RTD Instrumentation, °F;

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

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, values specified in the COLR;

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

τ_3 = Time constants utilized in the lag compensator for ΔT, value specified in the COLR;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER, °F;

K_1 = Value specified in the COLR;

K_2 = Value specified in the COLR;

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

τ_4, τ_5 = Time constants utilized in lead-lag compensator for T_{avg}, values specified in the COLR;

T = Measured RCS Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, value specified in the COLR;

TABLE NOTATIONS

NOTE 1: (Continued)

T' Indicated RCS T_{avg} at RATED THERMAL POWER, °F, (Calibration temperature for ΔT instrumentation, value specified in the COLR);

K_3 = Value specified in COLR;

P = Measured Pressurizer pressure, psig;

P' = Nominal RCS operating pressure, psig, value specified in the COLR;

S = Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers as specified in the COLR.

NOTE 2: Cycle dependent values for the channel's Allowable Value are specified in the COLR.

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{(1)}{(1+\tau_3 S)} \leq \Delta T_0 \{K_4 - K_5 \frac{(\tau_7 S)}{(1+\tau_7 S)} \frac{(1)}{(1+\tau_6 S)} T - K_6 [T \frac{(1)}{(1+\tau_6 S)} - T''] - f_2(\Delta I)\}$$

Where: ΔT = As defined in Note 1,

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1+\tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 = Value specified in the COLR,

K_5 = Value specified in the COLR,

$\frac{\tau_7 S}{1+\tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in rate-lag compensator for T_{avg}, value specified in the COLR,

$\frac{1}{1+\tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	Value specified in COLR,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER, °F, (Calibration temperature for ΔT instrumentation, value specified in the COLR),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	A function of the indicated difference between the top and bottom detectors of the power-range neutron ion chambers as specified in the COLR.

NOTE 4: Cycle dependent values for the channel's Allowable Value are specified in the COLR.

NOTE 5: Time constants utilized in the lead-lag controller for Pressurizer-Low are $\tau_1 \geq 10$ seconds and $\tau_2 \leq 1$ second.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2.

3.0.2 Upon discovery of a failure to meet an LCO, the ACTIONS shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified time interval, completion of the ACTION(S) is not required unless otherwise stated.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

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

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

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

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

This specification is not applicable in MODE 5 or 6.

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

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

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

APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.4 (Continued)

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

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

3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to Specifications 3.0.1 and 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the ACTIONS associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 6.7.6.o, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate ACTIONS of the LCO in which the loss of safety function exists are required to be entered.

When a support system's ACTION directs a supported system to be declared inoperable or directs entry into the ACTIONS for a supported system, the applicable ACTIONS shall be entered in accordance with LCO 3.0.2.

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 surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified 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.

SURVEILLANCE REQUIREMENTS

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

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

4.0.4 Entry into a MODE or other specified condition in the Applicability of a Limiting Condition for Operation (LCO) shall only be made when the LCO's Surveillances have been met within their specified frequency, except as provided by Specification 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 Specification 3.0.4.

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

4.0.5 Surveillance Requirements for inservice inspection of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code including applicable Addenda for the inservice inspection activities required by the ASME Boiler and Pressure Vessel Code including applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code including applicable Addenda terminology for inservice inspection activities	Required frequencies for performing service Inspection activities.
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Semi-quarterly	At least once per 46 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

4.0.5 (Continued)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection activities;

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

e. Deleted.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the SHUTDOWN MARGIN less than the limiting value, immediately initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limiting value:

a. DELETED

b. DELETED

c. When in MODE 2 with k_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 (Continued)

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

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

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the COLR. Additionally, the Reactor Coolant System boron concentration shall be greater than or equal to the limit specified in the COLR when the reactor coolant loops are in a drained condition.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than the limit specified in the COLR or the Reactor Coolant System boron concentration less than the limit specified in the COLR, immediately initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN and boron concentration are restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limit specified in the COLR and the Reactor Coolant System boron concentration shall be determined to be greater than or equal to the limit specified in the COLR when the reactor coolant loops are in a drained condition:

a. DELETED

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

    1) Reactor Coolant System boron concentration,

    2) Control rod position,

    3) Reactor Coolant System average temperature,

    4) Fuel burnup based on gross thermal energy generation,

    5) Xenon concentration, and

    6) Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: Beginning of cycle life (BOL) limit - MODES 1 and 2* only**.
End of cycle life (EOL) limit - MODES 1, 2, and 3 only**.

ACTION:

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

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR, within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

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

   3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.8.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

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

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

**See Special Test Exceptions Specification 3.10.3.

REACTIVITY CONTROL SYSTEMS

BORATION CONTROL

MODERATOR TEMPERATURE COEFFICIENT

SURVEILLANCE REQUIREMENTS

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

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

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

*Measurement of the MTC in accordance with Surveillance Requirement 4.1.1.3.b may be suspended provided that the benchmark criteria in WCAP-13749-P-A and the Revised Prediction specified in the COLR are satisfied.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2* **.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

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

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

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

**See Special Test Exceptions Specification 3.10.3.

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3/4.1.2 BORATION SYSTEMS

ISOLATION OF UNBORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 Provisions to isolate the Reactor Coolant System from unborated water sources shall be OPERABLE with:

a. The Boron Thermal Regeneration System (BTRS) isolated from the Reactor Coolant System, and

b. The Reactor Makeup Systems inoperable except for the capability of delivering up to the capacity of one Reactor Makeup Water pump to the Reactor Coolant System.

APPLICABILITY: MODES 4, 5, and 6

ACTION:

With the requirements of the above specification not satisfied immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and, if within 1 hour the required SHUTDOWN MARGIN is not verified, initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limits specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN is restored and the isolation provisions are restored to OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The provisions to isolate the Reactor Coolant System from unborated water sources shall be determined to be OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that at least the BTRS outlet valve, CS-V-302, or the BTRS moderating heat exchanger outlet valve, CS-V-305, or the manual outlet isolation valve for each demineralizer* not saturated with boron, CS-V-284, CS-V-295, CS-V-288, CS-V-290, CS-V-291, is closed and locked closed, and

b. Verifying that power is removed from at least one of the Reactor Makeup Water pumps, RMW-P-16A or RMW-P-16B.

*A demineralizer may be unisolated to saturate a bed with boron provided the effluent is not directed back to the Reactor Coolant System.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

b. With one full-length rod misaligned from its group step counter demand height by more than ± 12 steps, POWER OPERATION may continue provided that within 1 hour:

1. The remainder of the rods in the group with the misaligned rod are aligned to within ± 12 steps of the misaligned rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

2. The SHUTDOWN MARGIN is within the limits specified in the COLR. POWER OPERATION may then continue provided that:

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

b) The SHUTDOWN MARGIN is verified within the limits specified in the COLR at least once per 12 hours;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

3.1.3.1 ACTION b.2 (Continued)

c) A power distribution map is obtained from the Incore Detector System and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

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

c. DELETED

d. With more than one rod misaligned from its group step counter demand height by more than ± 12 steps:

1. Within 1 hour, verify SHUTDOWN MARGIN is within the limits specified in the COLR, and
2. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

---NOTE---

Surveillance Requirement 4.1.3.1.1 is not required to be performed for rods associated with inoperable digital rod position indicator or demand position indicator.

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions in accordance with the Surveillance Frequency Control Program.

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

TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)

Major Secondary Coolant System Pipe Rupture

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

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one DRPI per group inoperable, in one or more groups, either:

   1. Determine the position of the nonindicating rod(s) indirectly by the Incore Detector System at least once per 8 hours, or

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

b. With more than one DRPI per group inoperable in one or more groups:

   1. Immediately place the control rods in manual control,

   2. Verify the position of the rods with inoperable DRPIs using the Incore Detector System once per 8 hours, or reduce THERMAL POWER to less than 50% RATED THERMAL POWER, and

   3. Within 24 hours, restore inoperable DRPIs to OPERABLE status such that a maximum of one DRPI per group is inoperable, or be in MODE 3 in 6 hours.

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

   1. Verify that the position of the rods with inoperable DRPI using the Incore Detector System within 4 hours, or

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

LIMITING CONDITION FOR OPERATION (Continued)

d. With one or more demand position indicator(s) per bank inoperable in one or more banks, either:

   1. Verify that all DRPIs for the affected banks are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

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

SURVEILLANCE REQUIREMENTS

---NOTE---

Surveillance Requirement 4.1.3.2 is not required to be met for DRPIs associated with rods that do not meet Specification 3.1.3.1.

4.1.3.2 Each of the required DRPI(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full range of rod travel in accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

3/4.1.3.3 THIS SPECIFICATION NUMBER IS NOT USED

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the mechanical fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} for each loop greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System that could affect the drop time of those specific rods, and

c. In accordance with the Surveillance Frequency Control Program.

SHUTDOWN BANK INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown banks shall be fully withdrawn as specified in the COLR.

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

Not applicable to shutdown banks inserted while performing Surveillance Requirement 4.1.3.1.2.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

a. With one or more shutdown banks not fully withdrawn for reasons other than ACTION b:

1. Within 1 hour, verify SHUTDOWN MARGIN is within the limits specified in the COLR, or initiate boration to restore SHUTDOWN MARGIN to within limit, and
2. Within 2 hours, restore the shutdown banks to fully withdrawn as specified in the COLR or be in HOT STANDBY within the next 6 hours.

b. With one shutdown bank inserted ≤ 10 steps beyond fully withdrawn as specified in the COLR:

1. Within 1 hour, verify all control banks are within the insertion limits specified in the COLR,
2. Within 1 hour, verify SHUTDOWN MARGIN is within the limits specified in the COLR, or initiate boration to restore SHUTDOWN MARGIN to within limit, and
3. Within 24 hours, restore the shutdown bank to fully withdrawn as specified in the COLR, or be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and
b. In accordance with the Surveillance Frequency Control Program thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the COLR.

------------------------------NOTE----------------------------
Not applicable to control banks inserted while performing Surveillance Requirement 4.1.3.1.2.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

a. With the control banks inserted beyond the insertion limits specified in the COLR for reasons other than ACTION b:

   1. Within 1 hour, verify SHUTDOWN MARGIN is within the limits specified in the COLR, or initiate boration to restore SHUTDOWN MARGIN to within limit, and
   2. Restore the control banks to within the limits within 2 hours, or
   3. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or
   4. Be in at least HOT STANDBY within 6 hours.

b. With control bank A, B, or C inserted ≤ 10 steps beyond the insertion limit specified in the COLR:

   1. Within 1 hour, verify all shutdown banks are fully withdrawn as specified in the COLR,
   2. Within 1 hour, verify SHUTDOWN MARGIN is within the limits specified in the COLR, or initiate boration to restore SHUTDOWN MARGIN to within limit, and
   3. Within 24 hours, restore the control bank to within the insertion limits specified in the COLR, or be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits in accordance with the Surveillance Frequency Control Program.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER.

ACTION:

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

   1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

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

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

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.1.2 (THIS SPECIFICATION NUMBER IS NOT USED)

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3/4 2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$, as approximated by $F_Q^W(Z)$, shall be within the limits specified in the COLR:

APPLICABILITY: MODE 1.

ACTION:

With $F_Q^W(Z)$ exceeding its limit:

1. Within 4 hours, implement a RAOC operating space if specified in the COLR that restores $F_Q^W(Z)$ to within limits, and within 72 hours, perform SR 4.2.2.2.a and SR 4.2.2.2.b if control rod motion is required to comply with the new operating space. .

OR

2. Perform the following:

--------------- NOTE ----------------------
Required Action 2.4 shall be completed whenever Required Action 2.1 is performed prior to increasing THERMAL POWER above the limit of Required Action 2.1

   1. Within 4 hours, limit THERMAL POWER to less than RATED THERMAL POWER and reduce AFD limits as specified in the COLR, and

   2. Within 72 hours, reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER required by Action 2.1, and

   3. Within 72 hours, reduce Overpower ΔT trip setpoints ≥ 1% for each 1% that THERMAL POWER is limited below RATED THERMAL POWER required by Action 2.1, and

   4. Perform SR 4.2.2.2.a and 4.2.2.2.b prior to increasing THERMAL POWER above the limit of Required Action 2.1.

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SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits by:

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

b. Satisfying the $F_Q(Z)$ relationships specified in the COLR.

c. DELETED.

d. Verifying $F_Q^W(Z)$ to be within its limits according to the following schedule,

1) Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^W(Z)$ was last determined*, or

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

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

e. DELETED.

f. DELETED.

SURVELLLANCE REQUIREMENTS

g. DELETED.

4.2.2.3 DELETED.

4.2.2.4 (THIS SPECIFICATION NUMBER IS NOT USED)

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3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours reduce the THERMAL POWER to the level where the LIMITING CONDITION FOR OPERATION is satisfied.

b. THERMAL POWER may be increased, provided $F^N_{\Delta H}$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be demonstrated to be within its limit prior to operation above 75% RATED THERMAL POWER after each fuel loading and in accordance with the Surveillance Frequency Control Program thereafter by:

a. Using the Incore Detector System to obtain a power distribution map at any THERMAL POWER greater than 5% RATED THERMAL POWER.

b. Using the measured value of $F^N_{\Delta H}$ which does not include an allowance for measurement uncertainty.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the Incore Detector System to confirm indicated QUADRANT POWER TILT RATIO in accordance with the Surveillance Frequency Control Program by either:

a. Using the four pairs of symmetric detector locations or

b. Using the Incore Detector System to monitor the QUADRANT POWER TILT RATIO subject to the requirements of Technical Requirement TR20-3.3.3.2.

*See Special Test Exceptions Specification 3.10.2

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg} is less than or equal to the limit specified in the COLR,

b. Pressurizer Pressure is greater than or equal to the limit specified in the COLR*, and

c. Reactor Coolant System Flow shall be:

1. $\geq$ 374,400 gpm**; and,

2. $\geq$ 383,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits in accordance with the Surveillance Frequency Control Program.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

4.2.5.3 The RCS total flow rate shall be determined by an approved method to be within its limit prior to operation above 95% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

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

**Thermal Design Flow. An allowance for measurement uncertainty shall be made when comparing measured flow to Thermal Design Flow.

***Minimum measured flow used in the Revised Thermal Design Procedure.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit in accordance with the Surveillance Frequency Control Program. Each verification shall include at least one train and one channel per function.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1# #, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. (NOT USED)					
5. Intermediate Range, Neutron Flux	2	1	2	1# #, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown	2	0	1	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	10
7. Overtemperature ΔT	4	2	3	1, 2	6A
8. Overpower ΔT	4	2	3	1, 2	6A
9. Pressurizer Pressure--Low	4	2	3	1**	6A
10. Pressurizer Pressure--High	4	2	3	1, 2	6A
11. Pressurizer Water Level--High	3	2	2	1**	6A

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Reactor Coolant Flow—Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6A
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6A
13. Steam Generator Water Level--Low--Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6A
14. Undervoltage--Reactor Coolant Pumps	4-2/bus	2-1/bus	2 on one bus	1**	6A
15. Underfrequency--Reactor Coolant Pumps	4-2/bus	2-1/bus	2 on one bus	1**	6A
16. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1***	6B
b. Turbine Stop Valve Closure	4	4	4	1***	11
17. Safety Injection Input from ESF	2	1	2	1, 2	7
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2#	8

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1	8
f. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
19. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	7
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

**Trip function automatically blocked or bypassed below the P-7 (At Power) Setpoint.

***Trip function automatically blocked below the P-9 (Reactor Trip/Turbine Trip Interlock) Setpoint.

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

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

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

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

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

b. The Minimum Channels OPERABLE requirement is met; however, one channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.1.1, and

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

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

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, suspend all operations involving positive reactivity changes and verify that valve RMW-V31 is closed and secured in position within the next hour.

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

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

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

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 7 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in a least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 12 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 9. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-2

(This table number is not used)

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	SFCP(13)	N.A.	1,2,3*,4*,5*
2. Power Range, Neutron Flux						
a. High Setpoint	SFCP	SFCP(2, 4), SFCP(3, 4), SFCP(4, 6), SFCP(4, 5)	SFCP	N.A.	N.A.	1, 2
b. Low Setpoint	SFCP	SFCP(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1, 2
4. (NOT USED)						
5. Intermediate Range, Neutron Flux	SFCP	SFCP(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	SFCP	SFCP(4, 5)	S/U(8), SFCP(9)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
8. Overpower ΔT	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	SFCP	SFCP(16)	SFCP	N.A.	N.A.	1
10. Pressurizer Pressure--High	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	SFCP	SFCP	SFCP	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	SFCP	SFCP	SFCP	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP	N.A.	1
15. Underfrequency - Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP	N.A.	1
16. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	SFCP	N.A.	S/U(8, 10)	N.A.	1
b. Turbine Stop Valve	N.A.	SFCP	N.A.	S/U(8, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	SFCP(4)	SFCP	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
Reactor Trip System Interlocks (Continued)						
e. Power Range Neutron Flux, P-10	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	SFCP	SFCP	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	SFCP(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	SFCP(7)	1, 2, 3*, 4*, 5*
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	SFCP(7, 14), SFCP(15)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be closed and the Control Rod Drive System is capable of rod withdrawal.

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

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

(1) If not performed in previous 92 days.

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

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 50% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purposes of this surveillance requirement, monthly shall mean at least once per 31 EFPD.

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

(5) Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated and compared to the initial curves. The plateau curves for the Intermediate Range and Power Range detectors are required to be measured or obtained within 24 hours after attaining 100% of RATED THERMAL POWER. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(7) Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(8) If not performed in previous 31 days.

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

(10) Setpoint verification is not applicable.

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

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(12) Number not used.

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

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

(15) Automatic undervoltage trip.

(16) CHANNEL CALIBRATION shall include verification that the time constants are adjusted to the prescribed values.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

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

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

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

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

   Where:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit in accordance with the Surveillance Frequency Control Program. Each verification shall include at least one train and one channel per function.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Phase "A" Isolation, Containment Ventilation Isolation, Emergency Feedwater, Service Water to Secondary Component Cooling Water Isolation, CBA Emergency Fan/Filter Actuation, and Latching Relay).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13
c. Containment Pressure--Hi-1	3	2	2	1, 2, 3	18
d. Pressurizer Pressure--Low	4	2	3	1, 2, 3#	18
e. Steam Line Pressure--Low	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	18

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13
c. Containment Pressure--Hi-3	4	2	3	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	17

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (continued)					
b. Phase "B" Isolation (continued)					
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13
3) Containment Pressure--Hi-3	4	2	3	1, 2, 3	15
c. Containment Ventilation Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	16
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	16
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
4) Containment On Line Purge Radioactivity-High	2	1	2	1, 2, 3, 4	16
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
2) System	2	1	2	1, 2, 3	21

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (continued)					
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
c. Containment Pressure--Hi-2	3	2	2	1, 2, 3	18
d. Steam Line Pressure-Low	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	18
e. Steam Generator Pressure - Negative Rate-High	3/steam line	2/steam line any steam line	2/steam line	3*	18
5. Turbine Trip					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen.	3/stm. gen.	1, 2	18
6. Feedwater Isolation					
a. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen.	3/stm. gen.	1, 2	18
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

[illegible]TY FEATURES ACTUATION S[illegible]

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Emergency Feedwater					
a. Manual Initiation					
(1) Motor driven pump	1	1	1	1, 2, 3	21
(2) Turbine driven pump	2	1	2	1, 2, 3	21
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
c. Stm. Gen. Water Level--Low-Low					
Start Motor-Driven Pump and Start Turbine - Driven Pump	4/stm. gen.	2/stm. gen.	3/stm. gen.	1, 2, 3	18
d. Safety Injection Start Motor-Driven Pump and Turbine-Driven Pump	See Item 1. above for all Safety Injection initiating functions and requirements.				
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9 for Loss-of-Offsite Power initiating functions and requirements.				
8. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	13

Correction letter of 12-21-95

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	15
Coincident With: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
9. Loss of Power (Start Emergency Feedwater)					
a. 4.16 kV Bus E5 and E6-Loss of Voltage	2/bus	2/bus	1/bus	1, 2, 3, 4	14
b. 4.16 kV Bus E5 and E6-Degraded Voltage Coincident with SI	2/bus	2/bus	1/bus	1, 2, 3, 4	14
	See Item 1. above for all Safety Injection initiating functions and requirements.				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	19
b. Reactor Trip, P-4	2	2	2	1, 2, 3	21
c. Steam Generator Water Level, P-14	4/stm. gen.	2/stm. gen.	3/stm. gen.	1, 2, 3	18

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

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

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

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 14 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

b. The Minimum Channels OPERABLE requirements is met; however, the inoperable channel may be bypased for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition within 72 hours and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1.

ACTION 16 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

ACTION STATEMENTS (Continued)

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

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

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generator, Phase "A" Isolation, Containment Ventilation Isolation, and Emergency Feedwater, Service Water to Secondary Component Cooling Water Isolation, CBA Emergency Fan/Filter Actuation, and Latching Relay).					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--Hi-1	4.2	0.71	1.67	≤ 4.3 psig	≤ 5.3 psig
d. Pressurizer Pressure--Low	N.A.	N.A.	N.A.	≥ 1800 psig	≥ 1786 psig
e. Steam Line Pressure--Low	13.1	10.71	1.63	≥ 585 psig	≥ 568 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--Hi-3	3.0	0.71	1.67	≤ 18.0 psig	≤ 18.7 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Phase "B" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure--Hi-3	3.0	0.71	1.67	≤ 18.0 psig	≤ 18.7 psig
c. Containment Ventilation Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) Containment On-Line Purge Radioactivity-High	N.A.	N.A.	N.A.	< 2 x Background	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4.	Steam Line Isolation					
a.	Manual Initiation (System)	N.A.	N.A.	N.A.	N.A.	N.A.
b.	Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c.	Containment Pressure--Hi-2	5.2	0.71	1.67	≤4.3 psig	≤5.3 psig
d.	Steam Line Pressure--Low	13.1	10.71	1.63	≥585 psig	≥568 psig*
e.	Steam Generator Pressure - Negative Rate--High	3.0	0.5	0	≤100 psi	≤123 psi**
5.	Turbine Trip					
a.	Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b.	Steam Generator Water Level--High-High (P-14)	N.A	N.A.	N.A.	≤90.8% of narrow range instrument span.	≤91.3% of narrow range instrument span.
6.	Feedwater Isolation					
a.	Steam Generator Water Level--Hi-Hi-(P-14)	N.A.	N.A.	N.A.	≤90.8% of narrow range instrument span.	≤91.3% of narrow range instrument span.
b.	Safety Injection	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Emergency Feedwater					
a. Manual Initiation					
(1) Motor driven pump	N.A.	N.A.	N.A.	N.A.	N.A.
(2) Turbine driven pump	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low Start Motor-Driven Pump and Start Turbine-Driven Pump	N.A.	N.A.	N.A.	≥20.0% of narrow range instrument span.	≥19.5% of narrow range instrument span.
d. Safety Injection Start Motor-Driven Pump and Turbine-Driven Pump	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9. for Loss-of-Offsite Power Setpoints and Allowable Values.				
8. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level –Low-Low	4.0*** 2.1****	1.0	2.8	120,478 gals.	≤121,521*** gals. ≥119,435**** gals.
Coincident With Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Loss of Power (Start Emergency Feedwater)					
a. 4.16 kV Bus E5 and E6 Loss of Voltage	N.A.	N.A.	N.A.	≥ 2975 volts with a ≤ 1.20 second time delay.	≥ 2908 volts with a ≤ 1.315 second time delay.
b. 4.16 kV Bus E5 and E6 Degraded Voltage	N.A.	N.A.	N.A.	≥ 3933 volts with a ≤ 6 second time delay.	≥ 3902 volts with a ≤ 6.72 second time delay.
Coincident with: Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1950 psig	≤ 1962 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds.

***Value specified applies when “as measured” Trip Setpoint is greater than the specified Trip Setpoint.

****Value specified applies when “as measured” Trip Setpoint is less than the specified Trip Setpoint.

TABLE 3.3-5

(This table number is not used)

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generator, Phase "A" Isolation, Containment Ventilation Isolation, and Emergency Feedwater, Service Water to Secondary Component Cooling Water Isolation, CBA Emergency Fan/Filter Actuation, and Latching Relay).								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
c. Containment Pressure-Hi-1	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	SFCP	SFCP(4)	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
C. Containment Pressure-Hi-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation (System)	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3
c. Containment Pressure-Hi-2	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	SFCP	SFCP(4)	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	SFCP	SFCP(4)	SFCP	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2
b. Steam Generator Water Level-High-High (P-14)	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2
6. Feedwater Isolation								
a. Steam Generator Water Level--High-High (P-14)	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
7. Emergency Feedwater								
a. Manual Initiation								
1) Motor-driven pump	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
2) Turbine-driven pump	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Emergency Feedwater (Continued)								
b. Automatic Actuation and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3
c. Steam Generator Water Level-Low-Low, Start Motor-Driven Pump and Turbine-Driven Pump	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection, Start Motor-Driven Pump and Turbine-Driven Pump	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9. for all Loss-of-Offsite Power Surveillance Requirements.							
8. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
b. RWST Level Low-Low Coincident With	N.A.	SFCP	SFCP	SFCP(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. Loss of Power (Start) Emergency Feedwater)								
a. 4.16 kV Bus E5 and E6 Loss of Voltage	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV Bus E5 and E6 Degraded Voltage Coincident With	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	1, 2, 3
c. Steam Generator Water Level, P-14	SFCP	SFCP	SFCP	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3

TABLE NOTATION

(1) Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(2) A DIGITAL CHANNEL OPERATIONAL TEST will be performed on this instrumentation.

(3) Setpoint verification is not applicable.

(4) CHANNEL CALIBRATION shall include verification that the time constants are adjusted to the prescribed values.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-6
RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Deleted					
2. Containment Ventilation Isolation					
a. Deleted					
b. Manipulator Crane Area Monitor	1	2	6#	**	23
3. Main Steam Line	1/steam line	1/steam line	1, 2, 3, 4	N.A.	27
4. Fuel Storage Pool Areas					
a. Fuel Storage Building Exhaust Monitor	N.A.	1	***	****	25
5. Control Room Isolation					
a. Air Intake-Radiation Level					
1) East Air Intake	1/intake	2/intake	All	****	24
2) West Air Intake	1/intake	2/intake	All	****	24
6. Primary Component Cooling Water					
a. Loop A	1	1	All	$\leq$ 2 x Background	28
b. Loop B	1	1	All	$\leq$ 2 x Background	28

TABLE NOTATIONS

** Two times background or 15 mR/hr, whichever is greater.
*** With irradiated fuel in the fuel storage pool areas.
**** Two times background or 100 CPM, whichever is greater.
# During CORE ALTERATIONS or movement of irradiated fuel within the containment.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 23 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment ventilation isolation valves are maintained closed.

ACTION 24 - With the number of OPERABLE channels one less than 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 25 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool areas.

ACTION 27 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

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

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 within 14 days following the event outlining the actions taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 28 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, collect grab samples daily from the Primary Component Cooling Water System and the Service Water System and analyze the radioactivity until the inoperable Channel(s) is restored to OPERABLE status.

TABLE 4.3-3

(THIS TABLE NUMBER IS NOT USED)

3.3.3.2 (THIS SPECIFICATION NUMBER IS NOT USED)

3.3.3.3 (THIS SPECIFICATION NUMBER IS NOT USED

TABLE 3.3-7

(THIS TABLE NUMBER IS NOT USED)

TABLE 4.3-4

(THIS TABLE NUMBER IS NOT USED)

3.3.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

TABLE 3.3-8

(THIS TABLE NUMBER IS NOT USED)

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown System transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With the number of OPERABLE remote shutdown monitoring channels less than the Total Number of Channels as required by Table 3.3-9, within 60 days restore the inoperable channel(s) to OPERABLE status or, pursuant to Specification 6.8.2, submit a Special Report that defines the corrective action to be taken.

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

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel in Table 3.3-9 shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by performance of a CHANNEL CHECK, and

b. In accordance with the Surveillance Frequency Control Program by performance of a CHANNEL CALIBRATION.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit listed in Table 3.3-9, including the actuated components, shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program.

TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Intermediate Range Neutron Flux	2	1
2. Source Range Neutron Flux	2	1
3. Reactor Coolant Temperature - Wide Range for Loops 1 and 4		
a. T_c	2	2
b. T_H	2	2
4. Pressurizer Pressure	2	2
5. Pressurizer Level	2	2
6. Steam Generator Pressure	1/stm. gen.	1/stm. gen.
7. Steam Generator Water Level	1/stm. gen.	1/stm. gen.
8. Steam Generator-Emergency Feedwater Flow Rate	1/stm. gen.	1/stm. gen.
9. Boric Acid Tank Level	1/tank	1/tank

TRANSFER SWITCHES/CONTROL CIRCUITS

1. Emergency Feedwater Pump Steam Supply Valves MS-V-393
2. Emergency Feedwater Pump Steam Supply Valves MS-V-394
3. Emergency Feedwater Pump Steam Supply Valves MS-V-395
4. Emergency Feedwater Pump FW-P-37B
5. Emergency Feedwater Recirculation Valve FW-V-346
6. Emergency Feedwater Recirculation Valve FW-V-347
7. SG A EFW Control Valve FW-FV-4214 A
8. SG A EFW Control Valve FW-FV-4214 B
9. SG B EFW Control Valve FW-FV-4224 A
10. SG B EFW Control Valve FW-FV-4224 B
11. SG C EFW Control Valve FW-FV-4234 A
12. SG C EFW Control Valve FW-FV-4234 B
13. SG D EFW Control Valve FW-FV-4244 A
14. SG D EFW Control Valve FW-FV-4244 B
15. SG A Atmospheric Relief Valve MS-PV-3001
16. SG B Atmospheric Relief Valve MS-PV-3002
17. SG C Atmospheric Relief Valve MS-PV-3003

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN SYSTEM

TRANSFER SWITCHES/CONTROL CIRCUITS

18. SG D Atmospheric Relief Valve MS-PV-3004
19. MS Isolation Valves MS-V-86/88/90/92
20. Not Used
21. Pressurizer Heaters, Group A
22. Pressurizer Heaters, Group B
23. Charging Pump CS-P-2A
24. Charging Pump CS-P-2B
25. Charging Pump Suction from RWST CS-LCV-112D
26. Charging Pump Suction from RWST CS-LCV-112E
27. Pressurizer Relief Valve (PORV) RC-PCV-456A
28. Pressurizer Relief Valve (PORV) RC-PCV-456B
29. PORV Block Valve RC-V-122
30. PORV Block Valve RC-V-124
31. High Pressure Injection SI-V-138
32. High Pressure Injection SI-V-139
33. VCT Discharge Isolation Valve CS-LCV-112B
34. VCT Discharge Isolation Valve CS-LCV-112C

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

....NOTE..

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

....

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.8.2 within the next 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation to OPERABLE status.

b. With the number of OPERABLE accident monitoring instrumentation channels except the containment POST-LOCA high range area monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE channels for the containment Post-LOCA high range area monitor less than required by the Minimum Channels OPERABLE requirements, initiate an alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.8.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by performance of a CHANNEL CHECK, and

b. In accordance with the Surveillance Frequency Control Program by performance of a CHANNEL CALIBRATION.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a. Normal Range	2	1
b. Extended Range	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	4	2
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	4	2
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Generator Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Reactor Coolant System Subcooling Margin Monitor	2	1
11. Containment Building Water Level	2	1
12. Core Exit Thermocouples	4/core quadrant	2/core quadrant
13. Containment Post-LOCA Area Monitor	2	1

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
14. Intermediate Range Neutron Flux	2	1
15. Intermediate Range Neutron Flux Rate	2	1
16. Containment Isolation Valve Position*	2/Penetration	1/Penetration
17. Containment Enclosure Negative Pressure	2	1
18. Condensate Storage Tank Water Level**	2	1
19. Reactor Vessel Level Indication System	2	1
20. Containment Hydrogen Concentration	2	1
21. Containment Sump Isolation Valve Position***	2 (1 per valve)	1

*Applies to penetrations with 2 active valves in series. These valves are moved to the closed position by automatic signals.

**Calculated on basis of pressure sensed at suction to the Emergency Feedwater Pumps.

***Applies to CBS-V8 and CBS-V14 open indication on UL indicators.

LIMITING CONDITION FOR OPERATION

3.3.3.7 (This specification number is not used.)

TABLE 3.3-11

(This table number is not used)

LIMITING CONDITION FOR OPERATION

3.3.3.8 (This specification number is not used).

LIMITING CONDITION FOR OPERATION

3.3.3.9 (THIS SPECIFICATION NUMBER IS NOT USED)

TABLE 3.3-12

(THIS TABLE NUMBER IS NOT USED)

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

(THIS TABLE NUMBER IS NOT USED)

TABLE 4.3-5

(THIS TABLE NUMBER IS NOT USED

TABLE 4.3-5 (Continued)

TABLE NOTATIONS

(THIS TABLE NUMBER IS NOT USED)

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The explosive gas monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification declare the channel inoperable and take the ACTION shown in Table 3.3-13.

b. With the number of OPERABLE explosive gas monitoring instrumentation channels less than the Minimum Channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within 30 days or, if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 to explain why this inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-6.

TABLE 3.3-13

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. RADIOACTIVE GAS WASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
Oxygen Monitor (Process)	1	**	34

TABLE 3.3-13 (Continued)

EXPLOSIVE GAS MONITORING INSTRUMENTATION

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TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* (NOT USED)

** During GASEOUS RADWASTE TREATMENT SYSTEM operation.

*** (NOT USED)

(NOT USED)

ACTION STATEMENTS

ACTION 32 - (NOT USED)

ACTION 33 - (NOT USED)

ACTION 34 - With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement, operation of this RADIOACTIVE GAS WASTE SYSTEM may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 35 - (NOT USED)

TABLE 4.3-6

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. RADIOACTIVE GAS WASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
Oxygen Monitor (Process)	SFCP	N.A.	SFCP(4)	SFCP	**

TABLE 4.3-6 (Continued)

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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TABLE 4.3-6 (Continued)

TABLE NOTATIONS

* (NOT USED)

** During RADIOACTIVE WASTE GAS SYSTEM operation.

*** (NOT USED)

**** (NOT USED)

\# (NOT USED)

(1) (NOT USED)

(2) (NOT USED)

(3) (NOT USED)

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

(5) (NOT USED)

3/4.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two reactor coolant loops shall be OPERABLE with two reactor coolant loops in operation when the Control Rod Drive System is capable of rod withdrawal and one reactor coolant loop in operation when the Control Rod Drive System is not capable of rod withdrawal.*

APPLICABILITY: MODE 3.

ACTION:

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

b. With only one reactor coolant loop in operation and the Control Rod Drive System is capable of rod withdrawal, within 1 hour return the required reactor coolant loop to operation or place the Control Rod Drive System in a condition incapable of rod withdrawal.

c. With no reactor coolant loop in operation, place the Control Rod Drive System in a condition incapable of rod withdrawal, and suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

*All reactor coolant pumps may be deenergized for up to 1 hour per 8 hour period provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability*.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 14% in accordance with the Surveillance Frequency Control Program.

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

*Not required to be performed until 24 hours after a required pump is not in operation.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

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

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,**

e. RHR Loop A, and

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

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

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

---

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary-side water level to be greater than or equal to 14% in accordance with the Surveillance Frequency Control Program.

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

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

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

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

b. The secondary-side water level of at least two steam generators shall be greater than 14%.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

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

SURVEILLANCE RQUIREMENTS

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

4.4.1.4.2.1 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

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

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

**Within ±1% following pressurizer Code safety valve testing.

REACTOR COOLANT SYSTEM

SAFETY VALVES

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting* of 2485 psig ± 3%.**

APPLICABILITY: MODES 1, 2, and 3#.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

**Within ±1% following pressurizer Code safety valve testing.

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% of pressurizer level (1656 cubic feet), and at least two groups of pressurizer heaters each having a capacity of at least 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With the pressurizer otherwise inoperable, fully insert all rods, place the Control Rod Drive System in a condition incapable of rod withdrawal, and be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters from the emergency power supply and measuring circuit current in accordance with the Surveillance Frequency Control Program.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4.

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

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

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

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

ACTION:

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

Separate action entry is allowed for each steam generator tube.

----

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

   1. Within 7 days, verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, or be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours, 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 steam generator tube inspection.

b. With steam generator tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following RCS leakage detection systems shall be OPERABLE:

a. One containment atmosphere radioactivity monitor (gaseous or particulate), and

b. The containment drainage sump level monitoring system.

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

ACTION:

a. With the containment drainage sump level monitoring system inoperable:

1. Perform surveillance requirement 4.4.6.2.1.d, RCS inventory balance at least once per 24 hours*, and

2. Restore the containment drainage sump level monitoring system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the required containment atmosphere radioactivity monitor inoperable:

1. Perform surveillance requirement 4.4.6.2.1.d, RCS inventory balance, at least once per 24 hours*, or analyze grab samples of the containment atmosphere at least once per 24 hours, and

2. Restore the required inoperable containment atmosphere radioactivity monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the containment drainage sump level monitoring system inoperable and the containment atmosphere particulate monitor inoperable:

1. Enter Action a, and

2. Analyze grab samples of the containment atmosphere at least once per 12 hours, and

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3. Restore either the containment drainage sump level monitoring system or the containment atmosphere particulate monitor to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Required containment atmosphere radioactivity monitor:

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

2. Performance of a DIGITAL CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program, and

3. Performance of a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

b. Containment Drainage Sump Level Monitoring System - performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary to secondary leakage through any one steam generator (SG),

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

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 psig ± 20 psig, and

f. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve.*

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

ACTION:

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

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

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

3.4.6.2

ACTION: (Continued)

-----NOTE-----

Enter applicable ACTIONS for systems made inoperable by an inoperable pressure isolation valve.

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

SURVEILLANCE REQUIREMENTS

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

a. Not Used

b. Not Used

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

d. Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program during steady-state operation, except that not more than 96 hours shall elapse between any two successive inventory balances; (1) (2)

e. Monitoring the Reactor Head Flange Leakoff System in accordance with the Surveillance Frequency Control Program, and

f. Verifying primary to secondary leakage is ≤ 150 gallons per day through any one SG in accordance with the Surveillance Frequency Control Program. (2)

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(1) Not applicable to primary to secondary leakage.

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

SURVEILLANCE REQUIREMENTS

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. In accordance with the Surveillance Frequency Control Program,

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

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

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

e. Testing in accordance with the INSERVICE TESTING PROGRAM.

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

* Not applicable to RHR Pumps 8A and 8B suction isolation valves.

3/4.4.7 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

b. Less than or equal to 100/Ē microCuries per gram of gross radioactivity.

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

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

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

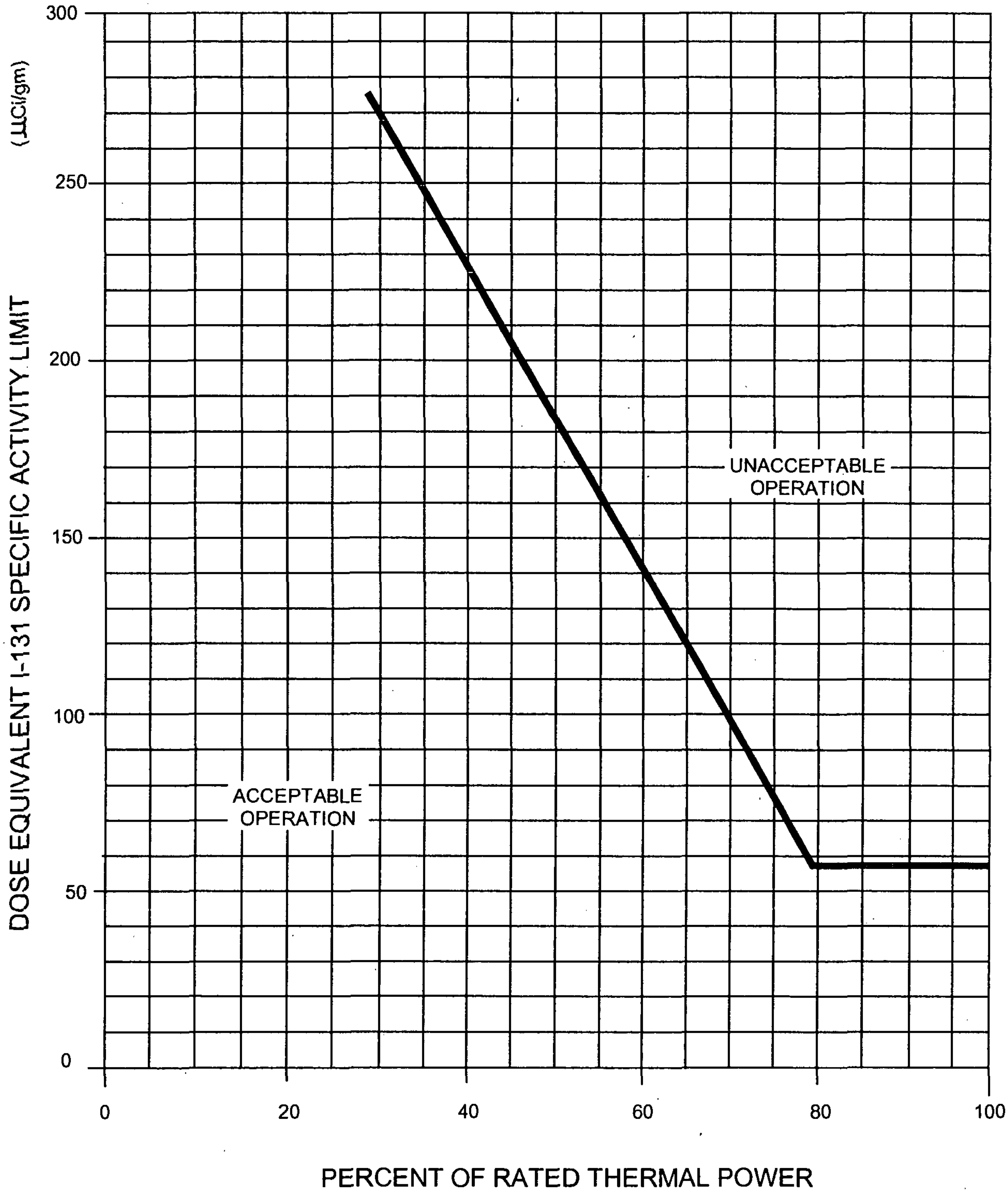
MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/Ē microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-3 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-3.

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



<u>FIGURE 3.4-1</u>

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 µCi/gram DOSE EQUIVALENT I-131

TABLE 4.4-3

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS

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

* A radiochemical analysis for Ē shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of Ē for the reactor coolant sample. Determination of the contributors to Ē shall be based upon those energy peaks identifiable with a 95% confidence level.

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

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

GENERAL

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIRMENTS

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

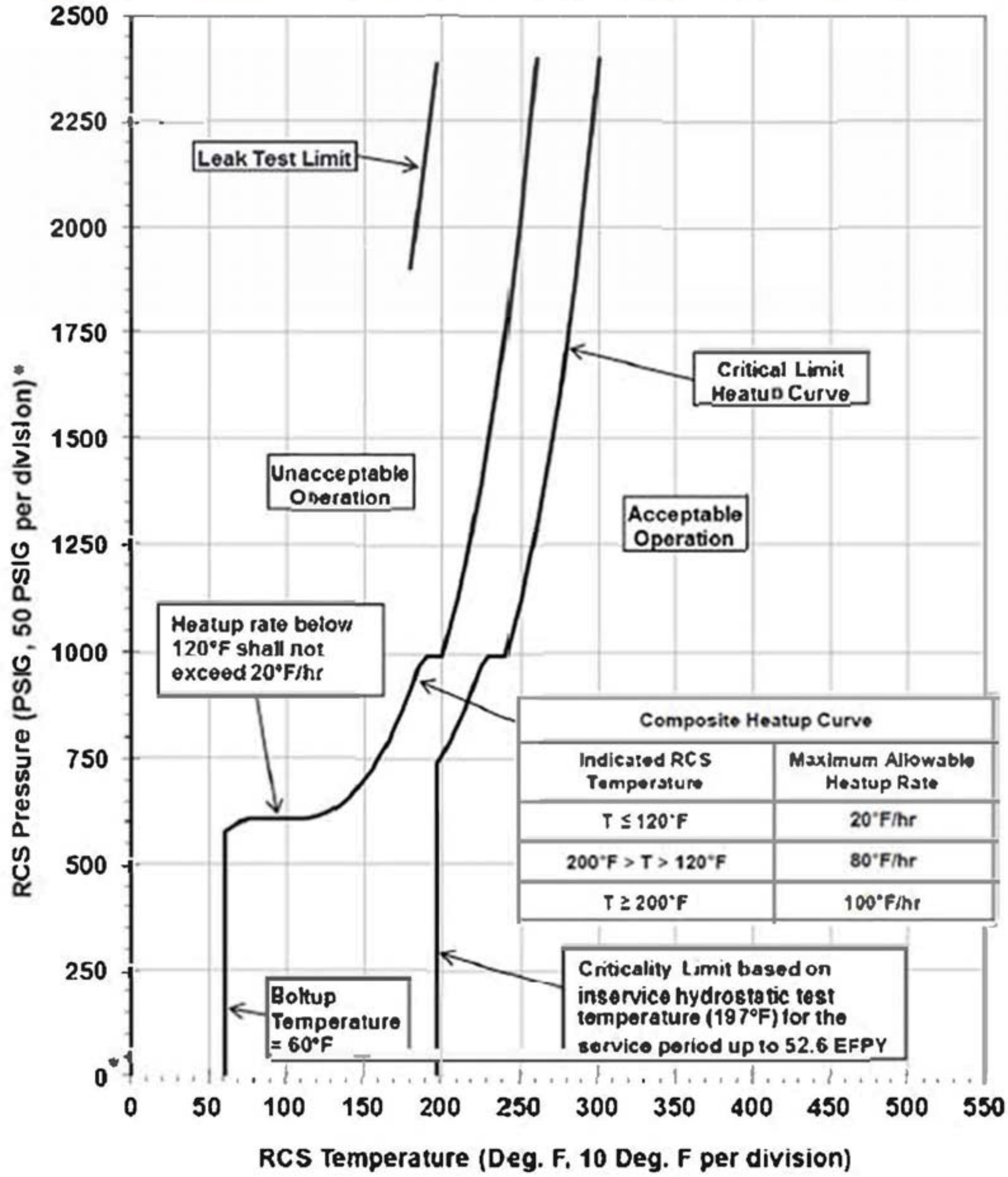
MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Plate R1808-1 without using surveillance data, Position 1.1

LIMITING ART VALUES AT 52.6 EFPY: 1/4T, 117°F (Axial Flaw)

3/4T, 105°F (Axial Flaw)

Curves applicable for the first 52.6 EFPY and contain margins for possible instrument errors



Composite Heatup Curve

Indicated RCS Temperature	Maximum Allowable Heatup Rate
T ≤ 120°F	20°F/hr
200°F > T > 120°F	80°F/hr
T ≥ 200°F	100°F/hr

* Curve is Applicable for RCS Vacuum fill.

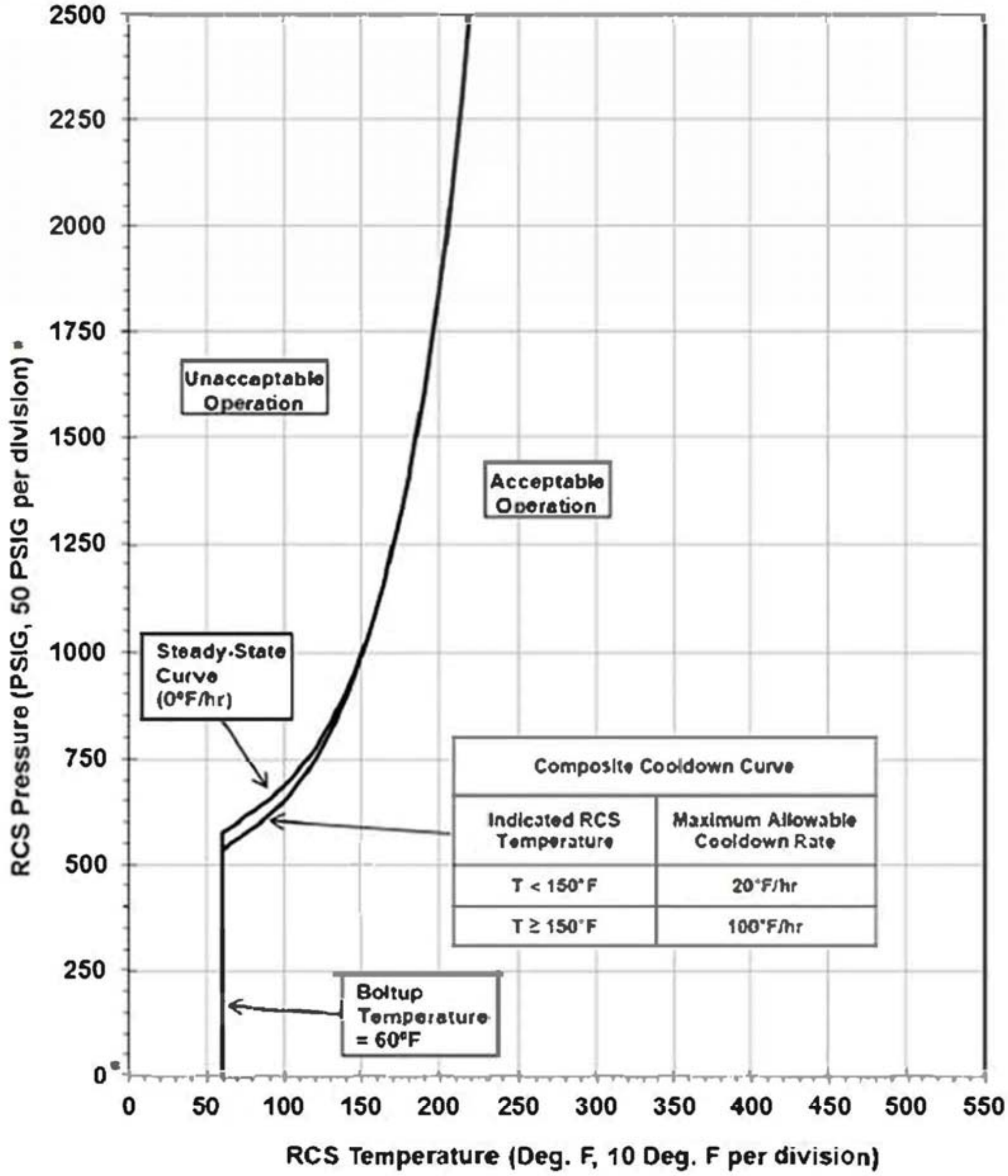
FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS – APPLICABLE UP TO 52.6 EFPY

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Plate R1808-1 without using surveillance data, Position 1.1
LIMITING ART VALUES AT 52.6 EFPY: 1/4T, 117°F (Axial Flaw)
3/4T, 105°F (Axial Flaw)

Curves applicable for the first 52.6 EFPY and contain margins for possible instrument errors



* Curve is Applicable for RCS Vacuum fill.

FIGURE 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS – APPLICABLE UP TO 52.6 EFPY

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

b. A maximum cooldown of 200°F in any 1-hour period, and

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. In MODE 4 when the temperature of any RCS cold leg is less than or equal to 225°F; and in MODE 5 and MODE 6 with all Safety Injection pumps inoperable at least one of the following groups of two overpressure protection devices shall be OPERABLE when the RCS is not depressurized with an RCS vent area of greater than or equal to 1.58 square inches:

   1) Two residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig +0, -3 %; or

   2) Two power-operated relief valves (PORVs) with lift setpoints that vary with RCS temperature which do not exceed the limit established in Figure 3.4-4, or

   3) One RHR suction relief valve and one PORV with setpoints as required above.

b. In MODE 5 and MODE 6 with all Safety Injection pumps except one inoperable:

   1) The Reactor Coolant System (RCS) depressurized with an RCS vent area equal to or greater than 18 square inches, or

   2) The RCS in a reduced inventory condition*.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 225°F; MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned.

*A reduced inventory condition exists whenever reactor vessel (RV) water level is lower than 36 inches below the RV flange.

3.4.9.3

ACTION: (Continued)

b) In MODE 5 and MODE 6 with all Safety Injection pumps inoperable and with one of the two required overpressure protection devices inoperable, restore two overpressure protection devices to OPERABLE status within 24 hours or within the next 8 hours

(a) depressurize the RCS and

(b) vent the RCS through at least a 1.58-square-inch vent.

c) In MODE 4, MODE 5 and MODE 6 with all Safety Injection pumps inoperable and with both of the two required overpressure protection devices inoperable, within the next 8 hours

(a) depressurize the RCS and

(b) vent the RCS through at least a 1.58-square-inch vent.

d) In the event the PORVs, or the RHR suction relief valves, 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 6.8.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e) In MODE 5 and MODE 6 with all Safety Injection pumps except one inoperable and with the RCS vent area less than 18 square inches or RCS water level not in a reduced inventory condition, immediately restore all Safety Injection pumps to inoperable status.

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3

ACTION: (Continued)

f) With more than one charging pump capable of injecting into the RCS, immediately initiate action to restore a maximum of one charging pump capable of injecting into the RCS.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORV(s) are being used for overpressure protection by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, in accordance with the Surveillance Frequency Control Program thereafter when the PORV is required OPERABLE; and

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

c. Verifying the PORV isolation valve is open in accordance with the Surveillance Frequency Control Program.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valve(s) are being used for overpressure protection as follows:

a. For RHR suction relief valve RC-V89 by verifying in accordance with the Surveillance Frequency Control Program that RHR suction isolation valves RC-V87 and RC-V88 are open.

b. For RHR suction relief valve RC-V24 by verifying in accordance with the Surveillance Frequency Control Program that RHR suction isolation valves RC-V22 and RC-V23 are open.

c. Testing in accordance with the INSERVICE TESTING PROGRAM.

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

**Except when the vent pathway is provided with a valve(s) or device(s) that is locked, sealed, or otherwise secured in the open position, then verify this valve(s) or device(s) open in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

4.4.9.3.4 The reactor vessel water level shall be verified to be lower than 36 inches below the reactor vessel flange in accordance with the Surveillance Frequency Control Program when the reduced inventory condition is being used for overpressure protection.

4.4.9.3.5 All charging pumps, excluding one OPERABLE pump, shall be demonstrated inoperable*** by verifying that the motor circuit breakers are secured in the open position**** in accordance with the Surveillance Frequency Control Program, except when the reactor vessel head closure bolts are fully detensioned or the vessel head is removed.

*** An additional pump may be made capable of injecting under administrative control for up to 1 hour during pump-swap operation, except during RCS water-solid conditions. Additionally, an inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

**** An alternate method to assure pump inoperability may be used by placing the control room pump-control switch in the Pull-to-Lock position and isolating the discharge flow path of the pump from the RCS by a least one closed isolation valve. Use of the alternative method requires inoperability verification in accordance with the Surveillance Frequency Control Program.

VALID FOR THE FIRST 52.6 EFPY, MAXIMUM SETPOINT ACCOUNTS
FOR INSTRUMENT UNCERTAINTIES

$T \leq 75.0°F$, $P = 520.0$ PSIG;

$75.0°F < T \leq 125.0°F$, $P = 1.5*(T-75.0) + 520.0$ PSIG;
$125.0°F < T \leq 160.0°F$, $P = 3.71*(T-125.0) + 595.0$ PSIG;
$160.0°F < T \leq 350.0°F$, $P = 725.0$ PSIG

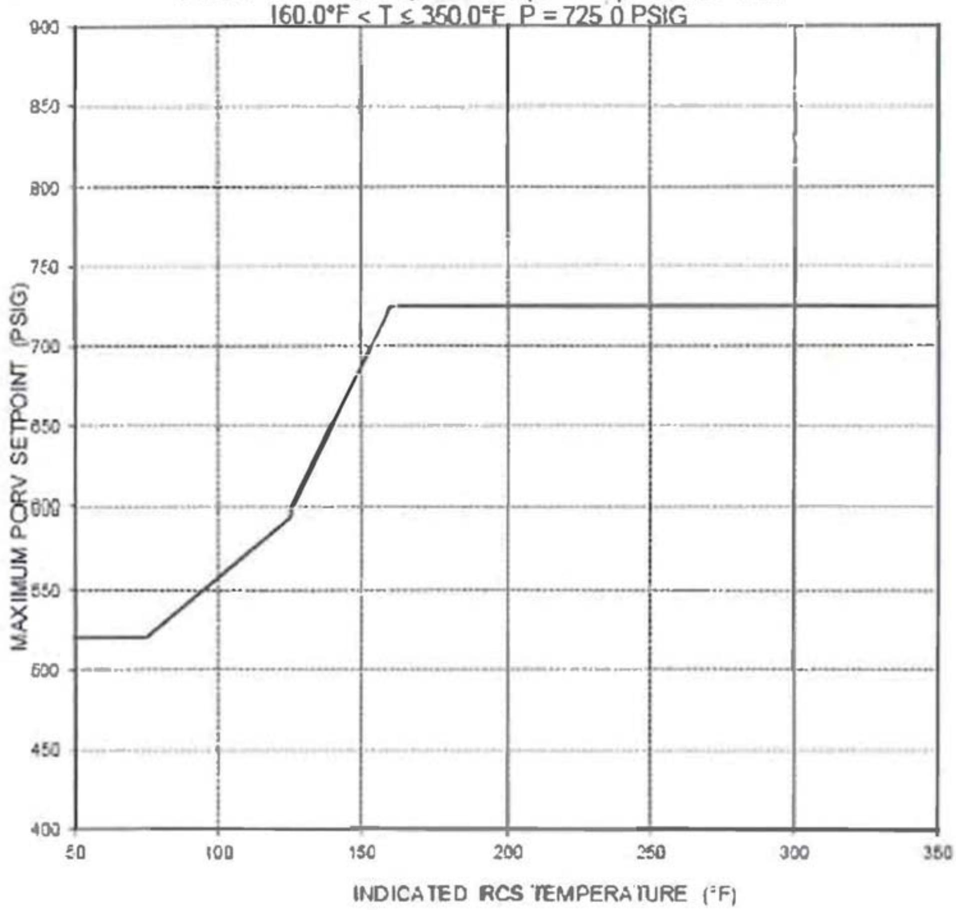


FIGURE 3.4-4 MAXIMUM ALLOWABLE PORV SETPOINTS FOR COLD OVERPRESSURE PROTECTION SYSTEM

*** Note that above the enable temperature the PORV setpoints will not restrict plant heatup and cooldown operations since COMS is not required to be armed at temperatures higher than 225°F. Hence the PORV setpoint values ramp up to the nominal setpoint value of 2385 psig is not shown.**

DELETED

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

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

a. Reactor vessel head, and

b. Pressurizer steam space.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each Reactor Coolant System vent path block valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per COLD SHUTDOWN, if not performed within the previous 92 days, by operating the valve through one complete cycle of full travel from the control room.

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

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

*For an OPERABLE vent path using a power-operated relief valve (PORV) as the vent path, the PORV block valve is not required to be closed.

SURVEILLANCE REQUIREMENTS

4.4.11.2 (Continued)

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

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

HOT STANDBY, STARTUP, AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6121 and 6596 gallons,

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

d. A nitrogen cover-pressure of between 585 and 664 psig.

APPLICABILITY: MODES 1, 2, and 3*

ACTION:

a. With one accumulator inoperable, except due to boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

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

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS

4.5.1.1 (Continued)

2) Verifying that each accumulator isolation valve is open.

b. By verifying the boron concentration of the accumulator solution under the following conditions:

1) In accordance with the Surveillance Frequency Control Program,

2) Within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume. This surveillance is not required when the volume increase makeup source is the RWST and the RWST has not been diluted since verifying that the RWST boron concentration is equal to or greater than the accumulator boron concentration limit.

c. In accordance with the Surveillance Frequency Control Program when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is disconnected.

EMERGENCY CORE COOLING SYSTEMS

ACCUMULATORS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each reactor coolant system accumulator isolation valve shall be shut with power removed from the valve operator.

APPLICABILITY: MODES 4* and 5**.

ACTION:

With one or more accumulator isolation valve(s) open and/or power available to the valve operator(s), immediately close the accumulator isolation valves and/or remove power from the valve operator(s).

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each accumulator isolation valve will be verified shut with power removed from the valve operator in accordance with the Surveillance Frequency Control Program.

*Within 12 hours prior to entry into MODE 3 from MODE 4 and if pressurizer pressure is greater than 1000 psig, each accumulator isolation valve shall be open as required by Specification 3.5.1.1.a.

**With accumulator pressure greater than 100 psig.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path* capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

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

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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*During MODE 3, the discharge paths of both Safety Injection pumps may be isolated by closing for a period of up to 2 hours to perform surveillance testing as required by Specification 4.4.6.2.2.

**The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump and the Safety Injection pumps declared inoperable pursuant to Specification 4.5.3.1.2 provided the centrifugal charging pump and the Safety Injection pumps are restored to OPERABLE status within at least 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

Valve Number	Valve Function	Valve Position
SI-V-114	SI Pump to Cold-Leg Isolation	Open
RH-V-14	RHR Pump to Cold-Leg Isolation	Open
RH-V-26	RHR Pump to Cold-Leg Isolation	Open
RH-V-32	RHR to Hot-Leg Isolation	Closed
RH-V-70	RHR to Hot-Leg Isolation	Closed
SI-V-77	SI to Hot-Leg Isolation	Closed
SI-V-102	SI to Hot-Leg Isolation	Closed

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

1) Verifying ECCS locations susceptible to gas accumulation are sufficiently filled with water, and

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

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

1) Verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 440 psig, the interlocks prevent the valves from being opened.

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

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

1) Verifying that each automatic valve in the flow path actuates to its correct position on (Safety Injection actuation and Automatic Switchover to Containment Sump) test signals, and

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

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying OPERABILITY of each pump when tested in accordance with the INSERVICE TESTING PROGRAM:

1) Centrifugal charging pump;

2) Safety Injection pump; and

3) RHR pump.

4.5.2 (Continued)

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

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

2) In accordance with the Surveillance Frequency Control Program.

High Head SI System Valve Number	Intermediate Head SI System Valve Number
SI-V-143	SI-V-80
SI-V-147	SI-V-85
SI-V-151	SI-V-104
SI-V-155	SI-V-109
	SI-V-117
	SI-V-121
	SI-V-125
	SI-V-129

h. NOT USED

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

----

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

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

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

SURVEILLANCE REQUIREMENTS

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

4.5.3.1.2 All centrifugal charging pumps and Safety Injection pumps, except the above allowed OPERABLE pumps, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position** within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first, and in accordance with the Surveillance Frequency Control Program thereafter.

*An additional charging pump may be made capable of injecting under administrative control for up to 1 hour during pump-swap operation, except during RCS water-solid conditions. Additionally, an inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

**An alternate method to assure pump inoperability may be used by placing the control room pump-control switch(s) in the Pull-to-Lock position and isolating the discharge flow path of the pump(s) from the RCS by at least one closed isolation valve. Use of the alternate method requires inoperability verification in accordance with the Surveillance Frequency Control Program.

ECCS SUBSYSTEMS - T_{avg} EQUAL TO OR LESS THAN 200°F

LIMITING CONDITION FOR OPERATION

3.5.3.2 As a minimum, the following number of Safety Injection pumps shall be inoperable*:

a. Two when the RCS vent area is less than 18 square inches.

b. One when the RCS vent area is equal to or greater than 18 square inches, or

c. One when the RCS is in a reduced inventory condition**.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned.

ACTION:

With fewer than the required number of Safety Injection pumps inoperable, immediately restore all pumps required to inoperable status.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps required to be inoperable shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position in accordance with the Surveillance Frequency Control Program***.

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

** A reduced inventory condition exists whenever reactor vessel (RV) water level is lower than 36 inches below the RV flange.

*** An alternate method to assure pump inoperability may be used by placing the control room pump-control switch(s) in the Pull-to-Lock position and isolating the discharge flow path of the pump(s) from the RCS by at least one closed isolation valve. Use of the alternate method requires inoperability verification in accordance with the Surveillance Frequency Control Program.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 477,000 gallons,

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

c. A minimum solution temperature of 50°F, and

d. A maximum solution temperature of 98°F.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

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

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. In accordance with the Surveillance Frequency Control Program by verifying the RWST temperature.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions except for valves that are open under administrative control as permitted by Specification 3.6.3; and

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

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

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the overall integrated leakage rate exceeding 1.0 La, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

CONTAINMENT AIR-LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

--------------------NOTE--------------------
Enter the ACTION of LCO 3.6.1.2, "Containment Leakage," when air lock leakage results in exceeding the overall containment leakage rate acceptance criteria.
----

a. With one containment air lock door inoperable:

   1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

   2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,

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

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

CONTAINMENT AIR-LOCKS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. With the leakage rate in accordance with the Containment Leakage Rate Testing Program.

b. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between 14.6 and 16.2 psia.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined in accordance with the Surveillance Frequency Control Program:

Location

a. Elevation 45 feet

b. Elevation 71 feet

c. Elevation 110 feet

d. Elevation 130 feet

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel in accordance with the Containment Leakage Rate Testing Program. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.8.2 within 15 days.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each 8-inch containment purge supply and exhaust isolation valve shall be OPERABLE and sealed closed except when open for purge system operation for pressure control; for ALARA, respirable, and air quality considerations to facilitate personnel entry; and for surveillance tests that require the valve(s) to be open.

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

ACTION:

a. With one or more of the 8-inch containment purge supply or exhaust isolation valves open for reasons other than given in Specification 3.6.1.7 above, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

b. With one or more containment purge supply or exhaust isolation valves having a measured leakage rate in excess of the limits of the Containment Leakage Rate Testing Program, restore the inoperable valve(s) to OPERABLE status or isolate the affected penetration(s) so that the measured leakage rate does not exceed the limits of the Containment Leakage Rate Testing Program, within 24 hours and close the purge supply if the affected penetration is the exhaust penetration, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT VENTILATION SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is in accordance with the Containment Leakage Rate Testing Program.

4.6.1.7.2 Each 8-inch containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7 in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST* and automatically transferring suction to the containment sump.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:
   1) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position**, and
   2) Verifying Containment Spray locations susceptible to gas accumulation are sufficiently filled with water.

b. By verifying OPERABILITY of each pump when tested in accordance with the INSERVICE TESTING PROGRAM;

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

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal, and
   2) Verifying that each spray pump starts automatically on a Containment Pressure-Hi-3 test signal.

d. By verifying each spray nozzle is unobstructed following activities that could result in nozzle blockage.

---

*In MODE 4, when the Residual Heat Removal System is in operation, an OPERABLE flow path is one that is capable of taking suction from the refueling water storage tank upon being manually realigned.

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

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 9420 and 9650 gallons of between 19 and 21% by weight NaOH solution, and

b. Two gravity feed paths each capable of adding NaOH solution from the chemical additive tank to the Refueling Water Storage Tank.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

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

   1) Verifying the contained solution volume in the tank, and

   2) Verifying the concentration of the NaOH solution by chemical analysis.

c. In accordance with the Surveillance Frequency Control Program by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE*.

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

ACTION:

---NOTES---

1. Enter applicable ACTIONS for systems made inoperable by containment isolation valves.

2. Enter the ACTION of LCO 3.6.1.2, "Containment Leakage," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

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

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

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

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

SURVEILLANCE REQUIREMENTS

4.6.3.1 Not used

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" Isolation valve actuates to its isolation position,

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" Isolation valve actuates to its isolation position, and

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*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

c. Verifying that on a Containment Purge and Exhaust Isolation test signal, each purge and exhaust valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic containment isolation valve shall be determined to be within its limit when tested in accordance with the INSERVICE TESTING PROGRAM.

3/4.6.4 COMBUSTIBLE GAS CONTROL

3.6.4.1 (THIS SPECIFICATION NUMBER IS NOT USED)

COMBUSTIBLE GAS CONTROL

3.6.4.2 (THIS SPECIFICATION NUMBER IS NOT USED)

HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 Two independent Containment Structure Recirculation Fan Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Containment Structure Recirculation Fan inoperable, restore the inoperable fan to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each Containment Structure Recirculation Fan System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by starting each system from the control room and verifying that the system operates for at least 15 minutes, and

b. In accordance with the Surveillance Frequency Control Program by verifying a system flow rate of at least 4000 cfm through the hydrogen mixing flow path.

3/4.6.5 CONTAINMENT ENCLOSURE BUILDING

CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two independent Containment Enclosure Emergency Air Cleanup System trains shall be OPERABLE.

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

ACTION:

With one Containment Enclosure Emergency Air Cleanup System train inoperable, restore the inoperable 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.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each Containment Enclosure Emergency Air Cleanup System train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

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

   1) Verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978*, and the system flow rate is 2100 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, by showing a methyl iodide penetration of less than or

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

SURVEILLANCE REQUIREMENTS

4.6.5.1b.2 (Continued)

equal to 5% when tested at a temperature of 30°C, at a relative humidity of 95% and a face velocity of 46 fpm in accordance with ASTM-D3803-1989; and

3) Verifying a system flow rate of 2100 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 5% when tested at a temperature of 30°C, at a relative humidity of 95% and a face velocity of 46 fpm in accordance with ASTM-D3803-1989.

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

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 2100 cfm ± 10%,

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

3) Verifying that the filter cross connect valves can be manually opened.

e. After each complete or partial replacement of a high efficiency particulate air (HEPA) filter bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a dioctyl phthalate (DOP) test aerosol while operating the system at a flow rate of 2100 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 2100 cfm ± 10%.

CONTAINMENT ENCLOSURE BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.2 Containment enclosure building integrity shall be maintained.

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

---NOTE---

Entry into ACTION is not required when the access opening is being used for normal transit entry or exit.

ACTION:

a. Without containment enclosure building integrity for reasons other than Action b, restore containment enclosure building integrity within 12 hours. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Without containment enclosure building integrity when equipment ingress and egress requires the access door to be maintained open, verify a dedicated individual, who is in continuous communication with the control room, is available to rapidly close the door; and restore containment enclosure building integrity 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

4.6.5.2 Containment enclosure building integrity shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that the door in each access opening is closed except when the access opening is being used for normal transit entry and exit, and

b. In accordance with the Surveillance Frequency Control Program by verifying the containment enclosure building can be maintained at a negative pressure greater than or equal to 0.25 inch water gauge by one train of the containment enclosure emergency air cleanup system within 4 minutes after a start signal.

CONTAINMENT SYSTEMS

CONTAINMENT ENCLOSURE BUILDING

CONTAINMENT ENCLOSURE BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.3 The structural integrity of the containment enclosure building shall be maintained at a level consistent with the Containment Leakage Rate Testing Program.

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

ACTION:

With the structural integrity of the containment enclosure building not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3 The structural integrity of the containment enclosure building shall be determined in accordance with the Containment Leakage Rate Testing Program. Any abnormal degradation of the containment enclosure building detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.8.2 within 15 days.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3#.

ACTION:

With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed, provided that within 4 hours either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR-LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	60
2	42
3	25

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					
Loop 1	Loop 2	Loop 3	Loop 4	LIFT SETTING* (± 3%)**	ORIFICE SIZE
V6	V22	V36	V50	1185 psig	16.0 sq. in.
V7	V23	V37	V51	1195 psig	16.0 sq. in.
V8	V24	V38	V52	1205 psig	16.0 sq. in.
V9	V25	V39	V53	1215 psig	16.0 sq. in.
V10	V26	V40	V54	1225 psig	16.0 sq. in.

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

**Within ±1% following main steam line Code safety valve testing.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. One motor-driven emergency feedwater pump, and one startup feedwater pump capable of being powered from an emergency bus and capable of being aligned to the dedicated water volume in the condensate storage tank, and

b. One steam turbine-driven emergency feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

---NOTE---

1. LCO 3.0.4.b is not applicable to the EFW pumps when entering MODE 1.
2. LCO 3.0.4.b is not applicable to the startup feedwater pump.

----

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two emergency feedwater pumps inoperable, restore at least one emergency feedwater pump to OPERABLE status within 12 hours and restore both emergency feedwater pumps to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one emergency feedwater pump and the startup feedwater pump inoperable, restore both emergency feedwater pumps to OPERABLE status within 24 hours and all three pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

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

1) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

2) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

3) Verifying that valves FW-156 and FW-163 are OPERABLE for alignment of the startup feedwater pump to the emergency feedwater header.

b. In accordance with the Surveillance Frequency Control Program by verifying the following pumps develop the required discharge pressure and flow as specified in the Technical Requirements Manual:

1) The motor-driven emergency feedwater pump;

2) The steam turbine-driven emergency feedwater pump when the secondary steam supply pressure is greater than 500 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

3) The startup feedwater pump.

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

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Emergency Feedwater System Actuation test signal;

2) Verifying that each emergency feedwater pump starts as designed automatically upon receipt of an Emergency Feedwater Actuation System test signal;

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 c. (Continued)

3) Verifying that with all manual actions, including power source and valve alignment, the startup feedwater pump starts within the required elapsed time; and

4) Verifying that each emergency feedwater control valve closes on receipt of a high flow test signal.

4.7.1.2.2 Auxiliary feedwater flow paths to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days, or after maintenance on an auxiliary feedwater pump that could have an effect upon pump performance, prior to entering MODE 2 by verifying normal flow to each steam generator from:

a. Each emergency feedwater pump, and

b. The startup feedwater pump via the main feedwater flow path and via the emergency feedwater header.

PLANT SYSTEMS

TURBINE CYCLE

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) system shall be OPERABLE with

a. A volume of 212,000 gallons of water contained in the condensate storage tank, and

b. A concrete CST enclosure that is capable of retaining 212,000 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST or the CST enclosure inoperable, within 4 hours restore the CST and the CST enclosure to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3a. The CST shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the contained water volume in the CST is within its limits.

b. The CST enclosure shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by an inspection to verify that CST enclosure integrity is maintained.

PLANT SYSTEMS

TURBINE CYCLE

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant shall be less than or equal to 0.1 μCi/gm DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the secondary coolant greater than 0.1 μCi/gm DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 In accordance with the Surveillance Frequency Control Program, verify the specific activity of the secondary coolant is less than or equal to 0.1 μCi/gm DOSE EQUIVALENT I-131.

*The provisions of Specification 4.0.4 are not applicable for entry into MODE 4, however, once steam generator pressure exceeds 100 psig, the requirements of Specification 4.7.1.4 must be met within 12 hours if not performed within the past 31 days.

TABLE 4.7-1

(THIS TABLE NUMBER IS NOT USED)

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3#.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5.0 seconds when tested in accordance with the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

PLANT SYSTEMS

TURBINE CYCLE

ATMOSPHERIC RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 At least four atmospheric relief valves and associated manual controls including the safety-related gas supply systems shall be OPERABLE.

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

ACTION:

a. With one less than the required atmospheric relief valves OPERABLE, restore the required atmospheric relief valves to OPERABLE status within 7 days; or be in at least HOT STANDBY within the next 12 hours.

b. With two less than the required atmospheric relief valves OPERABLE, restore at least three atmospheric relief valves to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric relief valve and associated manual controls including the safety-related gas supply systems shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the nitrogen accumulator tank is at a pressure greater than or equal to 500 psig.

b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, verify that all valves will open and close fully by operation of manual controls.

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*When steam generators are being used for decay heat removal.

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent primary component cooling water loops shall be OPERABLE, including one OPERABLE pump in each loop.

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

ACTION:

---NOTE---
Enter applicable ACTIONS of LCO 3.4.1.3, "Reactor Coolant Loops and Coolant Circulation," for residual heat removal loops made inoperable by PCCW.

With one primary component cooling water (PCCW) loop inoperable, restore the required primary component cooling water loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two primary component cooling water loops shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation signal.

3/4.7.4 SERVICE WATER SYSTEM/ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 The Service Water System shall be OPERABLE with:

a. An OPERABLE service water pumphouse and two service water loops with one OPERABLE service water pump in each loop,

b. An OPERABLE mechanical draft cooling tower and two cooling tower service water loops with one OPERABLE cooling tower service water pump in each loop, and

c. A portable cooling tower makeup system stored in its design operational readiness state.

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

ACTION:

---NOTES---

1. Enter applicable ACTIONS of LCO 3.8.1.1, "AC Sources- Operating," for diesel generator made inoperable by service water.

2. Enter applicable ACTIONS of LCO 3.4.1.3, "Reactor Coolant Loops and Coolant Circulation," for residual heat removal loops made inoperable by service water.

----

a. With one service water loop inoperable, return the loop to OPERABLE status within 72 hours, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one cooling tower service water loop or one cooling tower cell inoperable, return the affected loop or cell to OPERABLE status within 21 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two cooling tower service water loops or the mechanical draft cooling tower inoperable, return at least one loop and the mechanical draft cooling tower to OPERABLE status within 72 hours, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two loops (except as described in c) or the service water pumphouse inoperable, return at least one of the affected loops and the service water pumphouse to OPERABLE status within 24 hours, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

e. With the portable tower makeup pump system not stored in its design operational readiness state, restore the portable tower makeup pump system to its required condition within 72 hours, or continue operation and notify the NRC within the following 8 hours of actions to ensure an adequate supply of makeup water for the service water cooling tower for a minimum of 30 days.

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal.

4.7.4.2 Each service water cooling tower loop shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

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

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal,

   2) Each automatic valve in the flowpath actuates to its correct position on a Tower Actuation (TA) test signal and

   3) Each service water cooling tower pump starts automatically on a TA signal.

4.7.4.3 The service water pumphouse shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the water level to be at or above 25.1' (-15.9' Mean Sea Level).

SURVEILLANCE REQUIREMENTS

4.7.4.4 The mechanical draft cooling tower shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying the water in the mechanical draft cooling tower basin to be at a level of greater than or equal to 42.15* feet.

b. In accordance with the Surveillance Frequency Control Program by verifying that the water in the cooling tower basin to be at a bulk average temperature of less than or equal to 70°F.

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

1) Starting from the control room each cooling tower fan that is required to be OPERABLE and operating each of these fans for at least 15 minutes, and

2) Verifying that the portable tower makeup pump system is stored in its design operational readiness state.

d. In accordance with the Surveillance Frequency Control Program by verifying that the portable tower makeup pump develops a flow greater than or equal to 200 gpm.

*With the cooling tower in operation with valves aligned for tunnel heat treatment, the tower basin level shall be maintained at greater than or equal to 40.55 feet.

3/4.7.5 (THIS SPECIFICATION NUMBER IS NOT USED)

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3/4.7.6 CONTROL ROOM SUBSYSTEM

EMERGENCY MAKEUP AIR AND FILTRATION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent Control Room Emergency Makeup Air and Filtration System (CREMAFS) trains shall be OPERABLE.

---NOTE---

The control room envelope (CRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: All MODES
During movement of irradiated fuel assemblies

ACTION:

In MODE 1, 2, 3 or 4:

a. With one CREMAFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or both CREMAFS trains inoperable due to an inoperable CRE boundary:

   1. Immediately initiate action to implement mitigating actions or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

   2. Within 24 hours, verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

   3. Within 90 days, restore CRE boundary to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two CREMAFS trains inoperable for reasons other than an inoperable CRE boundary, immediately enter Technical Specification 3.0.3.

LIMITING CONDITION FOR OPERATION (Continued)

In MODE 5 or 6, or during movement of irradiated fuel assemblies:

d. With one CREMAFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable system to OPERABLE status within 7 days or either immediately initiate and maintain operation of the remaining OPERABLE CREMAFS train in the filtration/recirculation mode or immediately suspend movement of irradiated fuel assemblies.

e. With both CREMAFS trains inoperable, or with the OPERABLE CREMAFS train, required to be in the filtration/recirculation mode by ACTION d., not capable of being powered by an OPERABLE emergency power source, immediately suspend all movement of irradiated fuel assemblies.

f. With one or both CREMAFS trains inoperable due to an inoperable CRE boundary, immediately suspend movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each CREMAFS train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 continuous minutes with the heaters operating;

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

   1) Verifying that the filtration system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than .05% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978*, and the system flow rate is 1100 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, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 34.5 fpm (Train A) and 58.3 fpm (Train B) in accordance with ASTM-D-3803-1989;

   3) Verifying a system flow rate of 1100 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 34.5 fpm (Train A) and 58.3 fpm (Train B) in accordance with ASTM-D-3803-1989;

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

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks, for filter CBA-F-38, is less than 2.8 inches Water Gauge while operating the system at a flow rate of 1100 cfm ± 10%; and verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks, for filter CBA-F-8038, is less than 6.3 inches Water Gauge while operating the system at a flow rate of 1100 cfm ± 10%.

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

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that upon generation of an 'S' test signal, the following automatic system functions occur:

   a. The normal makeup air fan(s) trip off and the normal makeup air isolation damper(s) close;

   b. The control room exhaust subsystem isolation damper(s) close, and the exhaust fan trips off;

   c. The control room emergency makeup air and filtration subsystem actuates with flows through the HEPA filters and charcoal adsorber banks;

3) Verifying that upon generation of Remote Intake High Radiation test signal, the following automatic system functions occur:

   a. The normal makeup air fan(s) trip off and the normal makeup air isolation damper(s) close;

   b. The control room exhaust subsystem isolation damper(s) close, and the exhaust fan trips off;

   c. The control room emergency makeup air and filtration subsystem actuates with flows through the HEPA filters and charcoal adsorber banks;

4) Verifying that the heaters dissipate at least 3.24 kW (based on design rated voltage of 460V) when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the filtration system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than .05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1100 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the filtration system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than .05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 1100 cfm ± 10%.

g. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM SUBSYSTEMS

AIR CONDITIONING

LIMITING CONDITION FOR OPERATION

3.7.6.2 Two independent Control Room Air Conditioning Subsystems shall be OPERABLE.

APPLICABILITY: All MODES

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Air Conditioning Subsystem inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one Control Room Air Conditioning Subsystem inoperable, restore the inoperable system to OPERABLE status within 30 days or initiate and maintain operation of the remaining OPERABLE Control Room Air Conditioning Subsystem or immediately suspend all operations involving CORE ALTERATION.

b. With both Control Room Air Conditioning Subsystems inoperable, or with the OPERABLE Control Room Air Conditioning Subsystem unable to maintain temperature below the limiting equipment qualification temperature in the control room area, suspend all operations involving CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.7.6.2 Each Control Room Air Conditioning Subsystem shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the ability to maintain temperature in the control room area below the limiting equipment qualification temperature for 24 hours.

PLANT SYSTEMS

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation in accordance with the approved augmented inservice inspection program on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7 Each snubber shall be demonstrated OPERABLE by performance of the requirements of an approved augmented inservice inspection program.

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.8 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.8.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive materials:

   1. With a half-life greater than 30 days (excluding Hydrogen 3), and

   2. In any form other than gas.

SEALED SOURCE CONTAMINATION

SURVEILLANCE REQUIREMENTS

4.7.8.2 (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.8.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

3/4.7.9 (This specification number is not used)

3/4.7.10 (THIS SPECIFICATION NUMBER IS NOT USED)

TABLE 3.7-3

(THIS TABLE NUMBER IS NOT USED)

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System,

and

b. Two separate and independent diesel generators, each with:

1) A separate day fuel tank containing a minimum fuel volume fraction of 3/8 (600 gallons),

2) A separate Fuel Storage System containing a minimum volume of 62,000 gallons of fuel,

3) A separate fuel transfer pump,

4) Lubricating oil storage containing a minimum total volume of 275 gallons of lubricating oil, and

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

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

ACTION:

---------------------------------NOTE-----------------------------------
LCO 3.0.4.b is not applicable to the diesel generators.
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a. With an offsite circuit of the above required A.C. electrical power sources inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a for the OPERABLE offsite circuit within 1 hour and at least once per 8 hours thereafter;

2. Within 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s), declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable; and

3.8.1.1 (Continued)

ACTION:

3. Restore at least two offsite circuits to OPERABLE status within 72* hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* A one-time Allowed Outage Time (AOT) extension for an inoperable offsite circuit allows 240 hours to restore the inoperable Unit Auxiliary Transformers to OPERABLE status. Compensatory measures within NextEra Energy Seabrook, LLC letter L-2024-141, dated August 15, 2024, shall be implemented and shall remain in effect during the extended AOT period. The one-time AOT extension shall expire upon completion of the maintenance to restore the Unit Auxiliary Transformers to OPERABLE status or 240 hours, whichever occurs earliest. If the 240-hour one-time allowance has not been fully utilized, by November 30, 2024, this License Amendment will expire. MODE 1 operation is prohibited during the AOT.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 (Continued)

ACTION:

b. With a diesel generator inoperable:

1) Demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter. Perform ACTION d. Demonstrate the OPERABILITY of the remaining diesel generator by performing Specification 4.8.1.1.2a.5) within 24 hours.*

2) Restore at least two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, unless the following condition exists:

(a) The requirement for restoration of the diesel generator to OPERABLE status within 72 hours may be extended to 14** days if the Supplemental Emergency Power System (SEPS) is available, as specified in the Bases, and

(b) If at any time the SEPS availability cannot be met, either restore the SEPS to available status within 72 hours (not to exceed 14 days from the time the diesel generator originally became inoperable), or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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* The OPERABILITY of the remaining diesel generator need not be verified if it has been successfully operated within the last 24 hours, or if currently operating, or if the diesel generator became inoperable due to:

1. Preplanned preventive maintenance or testing,

2. An inoperable support system with no potential common mode failure for the remaining diesel generator, or

3. An independently testable component with no potential common mode failure for the remaining diesel generator.

**A one-time allowed outage time (AOT) extension for an inoperable diesel generator allows 30 days to restore the associated diesel generator to OPERABLE status. Compensatory measures within NEE Letters SBK-L-20068 dated July 13, 2020, and SBK-L-20117 dated September 23, 2020, will remain in effect during the extended AOT period. The one-time AOT extension shall expire upon completion of the maintenance or 90 days after the issuance of the amendment, whichever comes first. In addition, SEPS availability will be checked prior to entering the 30-day extended AOT, and subsequently once per shift during the 30-day extended AOT. If SEPS becomes unavailable any time during the extended AOT, restore SEPS to available within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. This 24-hour period will be allowed only once within any given extended EDG AOT.

LIMITING CONDITION FOR OPERATION

3.8.1.1 (Continued)

ACTION:

--NOTE--

Enter applicable ACTIONS of LCO 3.8.3.1, "Onsite Power Distribution – Operating," when ACTION c is entered with no AC power to any train.

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c. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable:

1) Demonstrate the OPERABILITY of the remaining A.C. source by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter. Perform ACTION d. Demonstrate the OPERABILITY of the remaining diesel generator by performing Specification 4.8.1.1.2a.5) within 8 hours.*

2) Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3) Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, unless the following condition exists:

(a) The requirement for restoration of the diesel generators to OPERABLE status within 72 hours may be extended to 14 days if the Supplemental emergency Power System (SEPS) is available, as specified in the Bases, and

(b) If at any time the SEPS availability cannot be met, either restore the SEPS to available status within 72 hours (not to exceed 14 days from the time the diesel generator originally became inoperable), or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* The OPERABILITY of the remaining diesel generator need not be verified if it has been successfully operated within the last 24 hours, or if currently operating, or if the diesel generator became inoperable due to:

1. Preplanned preventive maintenance or testing,

2. An inoperable support system with no potential common mode failure for the remaining diesel generator, or

3. An independently testable component with no potential common mode failure for the remaining diesel generator.

3.8.1.1 (Continued)

ACTION:

d. With one diesel generator inoperable in addition to ACTION b. or c. above, verify that:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and
2. When in MODE 1, 2, or 3, the steam-driven emergency feedwater pump is OPERABLE.

If these conditions are not satisfied within 4 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required offsite A.C. circuits inoperable

1. Within 12 hours from discovery of two offsite circuits inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) inoperable when its redundant required feature(s) is inoperable;
2. Restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours.
3. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With two of the above required diesel generators inoperable:

1) Demonstrate the OPERABILITY of two offsite A.C. circuits by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter.

2) Restore at least one diesel generator to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours,

3) Restore at least two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, unless the following condition exists:

(a) The requirement for restoration of the diesel generators to OPERABLE status within 72 hours may be extended to 14 days if the Supplemental Emergency Power System (SEPS) is available, as specified in the Bases, and

(b) If at any time the SEPS availability cannot be met, either restore the SEPS to available status within 72 hours (not to exceed 14 days from the time the diesel generator originally became inoperable), or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.*

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE.**

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

1) Verifying the fuel level in the day fuel tank;

2) Verifying the fuel level in the fuel storage tank;

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

4) Verifying the lubricating oil inventory in storage;

5) Verifying the diesel starts from standby conditions and attains a steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.***

* This surveillance requirement shall not be performed in Mode 1 or 2.

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

*** A modified start involving idling and gradual acceleration to synchronous speed may be used for this surveillance. When modified start procedures are not used, the time, voltage, and frequency tolerances of Specification 4.8.1.1.2e must be met.

4.8.1.1.2 (Continued)

6) Verifying the generator is synchronized, gradually loaded**** to greater than or equal to 5600 kW and less than or equal to 6100 kW, and operates within this load band for at least 60 minutes, and until stable engine operating temperature is attained; and

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

b. In accordance with the Surveillance Frequency Control Program by checking for and removing accumulated water from the day fuel tank;

c. In accordance with the Surveillance Frequency Control Program by checking for and removing accumulated water from the fuel oil storage tanks;

d. By verifying fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program;

e. In accordance with the Surveillance Frequency Control Program# by verifying the diesel starts from standby condition and achieves:

1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

2) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

**** Diesel generator loading may be in accordance with manufacturers recommendations, including a warmup period. Momentary transients outside the load range, due to changing bus conditions, do not invalidate the test. In addition, this surveillance shall be preceded by and immediately follow without shutdown a successful performance of Specification 4.8.1.1.2a.5) or 4.8.1.1.2e.

Performance of Specification 4.8.1.1.2a.6) must immediately follow this surveillance. Additionally, performance of Specification 4.8.1.1.2e satisfies Specification 4.8.1.1.2a.5).

4.8.1.1.2 (Continued)

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

1) (NOT USED)

2) Verifying the generator capability to reject a load of greater than or equal to 671 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 4.0 Hz;

3) Verifying the generator capability to reject a load of 6083 kW without tripping. The generator voltage shall not exceed 4992 volts during and following the load rejection;

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

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

b) Verifying the diesel starts from standby conditions### on the loss of offsite power signal, energizes the emergency buses with permanently connected loads within 12 seconds, energizes the auto-connected shutdown loads through the emergency power sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an SI actuation test signal, without loss-of-offsite power, the diesel generator starts from standby conditions### on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test;

Selected surveillance requirements, or portions thereof, may be performed during conditions or modes other than shutdown, provided an evaluation supports safe conduct of that surveillance in a condition or mode that is consistent with safe operation of the plant. (Ref. NRC GL 91-04)

Starting of the diesel for Specifications 4.8.1.1.2f.4) and 4.8.1.1.2f.5) may be performed with the engine at or near normal operating temperature.

4.8.1.1.2 (Continued)

6) Simulating a loss-of-offsite power in conjunction with an SI actuation test signal; and

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

   b) Verifying the diesel starts from standby conditions, on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the emergency power sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed, low lube oil pressure, 4160-volt bus fault, and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection actuation signal.

7) Verifying full-load carrying capability of the diesel generator for an interval of not less than 24 hours:

   a) At a load greater than or equal to 5600 kW and less than or equal to 6100 kW,#### or

   b) Should auto-connected loads be equal to or greater than 6100 kW;

      1. Verify the diesel generator operates for an interval of not less than 2 hours at a load greater than or equal to 6363 kW and less than or equal to 6700 kW.#### For the remaining hours, at a load greater than or equal to 5600 kW and less than or equal to 6100 kW, and

      2. Verify that the auto-connected loads to each diesel generator do not exceed the short time rating of 6700 kW.

Diesel generator loading may be in accordance with manufacturers recommendations, including a warmup period. Momentary transients outside the load range, due to changing bus conditions, do not invalidate the test.

4.8.1.1.2 (Continued)

8) Within 5 minutes of shutting down the diesel generator, after the diesel generator has operated for an interval of not less than 2 hours at a load greater than or equal to 5600 kW and less than or equal to 6100 kW,+ by verifying the diesel starts and achieves:

   a) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

   b) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

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

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

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

13) NOT USED

+ Momentary transients outside the load range, due to changing bus conditions, do not invalidate the test.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

14) Simulating a Tower Actuation (TA) signal while the diesel generator is loaded with the permanently connected loads and auto-connected emergency (accident) loads, and verifying that the service water pump automatically trips, and that the cooling tower pump automatically starts. After energization the steady state voltage and frequency of the emergency buses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz; and

15) While diesel generator 1A is loaded with the permanently connected loads and auto-connected emergency (accident) loads, manually connect the 1500 hp startup feedwater pump to 4160-volt bus E5. After energization the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz.

g. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously from standby condition and verifying that both diesel generators achieve:

1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

2) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

h. (THIS SPECIFICATION NUMBER IS NOT USED)

4.8.1.1.3 (THIS SPECIFICATION NUMBER IS NOT USED)

TABLE 4.8-1

(THIS TABLE NUMBER IS NOT USED)

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

1) A day fuel tank containing a minimum fuel volume fraction of 3/8 (600 gallons of fuel),

2) A fuel storage system containing a minimum volume of 60,000 gallons of fuel,

3) A fuel transfer pump, lubricating oil, and

4) Lubricating oil storage containing a minimum total volume of 275 gallons of lubricating oil, and

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

APPLICABILITY: MODES 5 and 6.

ACTION

---NOTE---

Enter the ACTION of LCO 3.8.3.2, "Onsite Power Distribution – Shutdown," with one required train de-energized as a result of inoperable offsite circuit.

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 1.58-square-inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1a, 4.8.1.1.2a [except for Specification 4.8.1.1.2a.6] and 4.8.1.1.2 b, c, d, e.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. Train A

1) 125-volt Battery Bank 1A or 1C,
2) One full-capacity battery charger on Bus #11A, and
3) One full-capacity battery charger on Bus #11C.

b. Train B

1) 125-volt Battery Bank 1B or 1D,
2) One full-capacity battery charger on Bus #11B, and
3) One full-capacity battery charger on Bus #11D.

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

ACTION:

a. With the required battery bank in one train inoperable, restore the battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the full-capacity chargers inoperable, restore the inoperable charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

1) The parameters in Table 4.8-2 meet the Category A limits, and
2) The total battery terminal voltage is greater than or equal to 128 volts on float charge.

b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

SURVEILLANCE REQUIREMENTS

4.8.2.1b (Continued)

1) The parameters in Table 4.8-2 meet the Category B limits,

2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x 10^{-6} ohm, and

3) The average electrolyte temperature of 16 connected cells (4 cells per row) is above 65°F.

c. In accordance with the Surveillance Frequency Control Program by verifying that:

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150 x 10^{-6} ohm, and

4) Each battery charger will supply at least 150 amperes at a minimum of 132 volts for at least 8 hours.

d. In accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. In accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

PARAMETER	CATEGORY A[1] LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B[2] LIMITS FOR EACH CONNECTED CELL	ALLOWABLE[3] VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[6]	> 2.07 volts
Specific Gravity[4]	≥ 1.200[5]	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[5]

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on float charge.

(6) Corrected for average electrolyte temperature.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt battery bank in one D.C. Train and the two associated full-capacity chargers shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With less than one battery bank and two chargers in one DC train OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required battery bank and full-capacity chargers to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a 1.58-square-inch vent.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above-required 125-volt battery bank and full-capacity chargers shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner:

a. Train A, A.C. Emergency Busses consisting of:

   1) 4160-volt Emergency Bus #E5,
   2) 480-volt Emergency Bus #E51,** and
   3) 480-volt Emergency Bus #E52.**

b. Train B, A.C. Emergency Busses consisting of:

   1) 4160-volt Emergency Bus #E6,
   2) 480-volt Emergency Bus #E61,**
   3) 480-volt Emergency Bus #E62,** and
   4) 480-volt Emergency Bus #E64.

c. 120-volt A.C. Vital Panel #1A energized from its associated inverter connected to D.C. Bus #11A,*

d. 120-volt A.C. Vital Panel #1B energized from its associated inverter connected to D.C. Bus #11B,*

e. 120-volt A.C. Vital Panel #1C energized from its associated inverter connected to D.C. Bus #11C,*

f. 120-volt A.C. Vital Panel #1D energized from its associated inverter connected to D.C. Bus #11D,*

g. 120-volt A.C. Vital Panel #1E energized from its associated inverter connected to D.C. Bus #11A,*

h. 120-volt A.C. Vital Panel #1F energized from its associated inverter connected to D.C. Bus #11B,*

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*Two inverters may be disconnected from their D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

**These busses can be considered OPERABLE if the 480 volt bus ties are closed. These bus ties will be under administrative control to ensure loading is within transformer rating.

LIMITING CONDITION FOR OPERATION

3.8.3.1 (Continued)

i. Train A, 125-volt D.C. Busses consisting of:

1) 125-volt D.C. Bus #11A energized from Battery Bank 1A or 1C, and
2) 125-volt D.C. Bus #11C energized from Battery Bank 1C or 1A.

j. Train B, 125-volt D.C. Busses consisting of:

1) 125-volt D.C. Bus #11B energized from Battery Bank 1B or 1D, and
2) 125-volt D.C. Bus #11D energized from Battery Bank 1D or 1B.

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

ACTION:

--NOTE--

Enter applicable ACTIONS of LCO 3.8.2.1, “DC Sources – Operating,” for DC trains made inoperable by inoperable AC power distribution system.

----

a. With one of the required trains of A.C. emergency busses (except 480-volt Emergency Bus # E64) not fully energized, reenergize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   1. With 480-volt Emergency bus #E64 not fully energized, reenergize the bus within 7 days or be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital panel within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panel from its associated inverter connected to its associated D.C. bus within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from an OPERABLE battery bank, reenergize the D.C. bus from an OPERABLE battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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LIMITING CONDITION FOR OPERATION

3.8.3.1 (Continued)

ACTION

d. With two or more A.C. vital panels of the same electrical train either not energized from their associated inverter, or with their inverters not connected to their associated D.C. bus: (1) reenergize or verify energized all A.C. vital panels within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize or verify energized at least two A.C. vital panels from their associated inverters connected to their associated D.C. bus within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses and panels shall be determined energized in the required manner in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical buses shall be energized in the specified manner:

a. One train of A.C. emergency buses consisting of the 4160-volt and the 480-volt A.C. emergency buses listed in 3.8.3.1a. and b. (excluding 480-volt Emergency Bus #E64);

b. Two of the four 120-volt A.C. vital Panels 1A, 1B, 1C, and 1D energized from their associated inverters connected to their respective D.C. buses;

c. One of the two 120-volt A.C. Vital Panels 1E or 1F energized from its associated inverter connected to the respective D.C. bus; and

d. Two 125-volt D.C. buses (in the same train) energized from the associated battery bank.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical buses and panels not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical buses and panels in the specified manner as soon as possible, and within 8 hours, depressurize and vent the RCS through at least a 1.58-square-inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified buses and panels shall be determined energized in the required manner in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

ONSITE POWER DISTRIBUTION

TRIP CIRCUIT FOR INVERTER I-2A

LIMITING CONDITION FOR OPERATION

3.8.3.3 The safety-related trip circuit that trips the D.C. feed from D.C. Bus #11C to inverter #I-2A after 15 minutes of discharge from the battery shall be OPERABLE. Note that this LIMITING CONDITION FOR OPERATION is applicable only when D.C. Bus #11C is required to be OPERABLE.

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

ACTION:

With this safety-related trip circuit inoperable, restore the trip circuit to OPERABLE status within 7 days or de-energize the D.C. feed to inverter #I-2A by tripping the D.C. circuit breaker in D.C. Bus #11C. Verify that this breaker is open once per 7 days thereafter.

SURVEILLANCE REQUIREMENTS

4.8.3.3 The safety-related trip circuit shall be demonstrated operable in accordance with the Surveillance Frequency Control Program.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 The circuit breakers feeding the following loads inside primary containment shall be padlocked in the open position:

Loads	Circuit	Panel
Refueling Canal Skimmer Pump	1-SF-P-272	1-ED-MCC-111
Polar Gantry Crane	1-MM-CR-3	1-ED-US-11
Distribution Panel	1-ED-PP-7A	1-ED-US-11
Distribution Panel	1-ED-PP-7B	1-ED-US-23
Rod Control Cluster Change Fixture	1-FH-RE 12	1-ED-MCC-111

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

EXCEPTION:

If any of the above-mentioned loads are required for brief durations (not to exceed 72 hours) during plant operation, the pertinent circuit breaker can be unlocked and closed for the required duration provided this change in breaker position becomes part of the applicable operating procedure used for the work inside containment.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Verify in accordance with the Surveillance Frequency Control Program that the circuit breakers listed above are locked in the open position.

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ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.3 Each thermal overload protection for safety-related motor-operated valves shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above-required valves inoperable, bypass the inoperable thermal overload within 8 hours, restore the inoperable thermal overload to OPERABLE status within 30 days, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for the above required valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program and following maintenance on the motor starter by selection of a representative sample of at least 25% of all thermal overloads for the above-required valves and replacing them with precalibrated devices that have been subjected to a CHANNEL CALIBRATION.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

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 uniform and sufficient to ensure a boron concentration of greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the boron concentration is restored to greater than or equal to the limit specified in the COLR.

SURVEILLANCE REQUIREMENTS

4.9.1.1 Verify boron concentration is within the limits specified in the COLR prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, immediately initiate corrective action to restore one source range neutron flux monitor to OPERABLE status and determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,

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

*Neutron detectors may be excluded from CHANNEL CALIBRATION.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 80 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 80 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 80 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, however both doors of one personnel airlock may be open if:

   1) One personnel airlock door is capable of being closed, and

   2) A designated individual is available outside the personnel airlock to close the door.

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 by an OPERABLE automatic Containment Purge and Exhaust Isolation System; or

   3) Be capable of being closed by a designated individual available at the penetration.*

APPLICABILITY: During movement of recently irradiated fuel within the containment.

ACTION:

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

* A designated individual shall not be used for manual isolation of valves CAP-V1, CAP-V2, CAP-V3, and/or CAP-V4.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

SURVEILLANCE REQUIREMENTS

4.9.4 For the above required containment building penetrations:

a. Determine that each of the above required containment building penetrations shall be in its required condition within 100 hours prior to the start of, and in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel in the containment building, and

b. Demonstrate that the Containment Purge and Exhaust Isolation System is OPERABLE in accordance with the Surveillance Frequency Control Program and within 10 days prior to the start of movement of recently irradiated fuel in the containment building by verifying that containment purge and exhaust isolation occurs on manual initiation and on a High Radiation test signal from each of the manipulator crane radiation area monitoring instrumentation channels. **

** Not required for those valves complying with Specification 3.9.4.c.1 or Specification 3.9.4.c.3.

3/4.9.5 (This specification number is not used.)

3/4.9.6 (This specification number is not used.)

3/4.9.7 (This specification number is not used.)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2750 gpm in accordance with the Surveillance Frequency Control Program.

4.9.8.1.1 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

* The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2750 gpm in accordance with the Surveillance Frequency Control Program.

4.9.8.2.1 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

* Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.9 (This specification number is not used.)

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and in accordance with the Surveillance Frequency Control Program thereafter during movement of fuel assemblies or control rods.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.12 FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent trains of the Fuel Storage Building Emergency Air Cleaning System shall be OPERABLE whenever irradiated fuel is in the storage pool and shall be OPERABLE with one train operating during fuel movement.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one train of the Fuel Storage Building Emergency Air Cleaning System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE train of the Fuel Storage Building Emergency Air System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no trains of the Fuel Storage Building Emergency Air Cleaning System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one train of the Fuel Storage Building Emergency Air Cleaning System is restored to OPERABLE status and is in operation.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required trains of the Fuel Storage Building Emergency Air Cleaning System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 continuous minutes with the heaters operating;

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

4.9.12b (Continued)

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 16,450 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, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 44 fpm in accordance with ASTM-D-3803-1989; and

3) Verifying a system flow rate of 16,450 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 44 fpm in accordance with ASTM-D-3803-1989.

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

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 16,450 cfm ± 10%,

2) Verifying that the system maintains the spent fuel storage pool area at a negative-pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation,

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

4.9.12d (Continued)

3) Verifying that the filter cross connect valve can be manually opened, and

4) Verifying that the heaters dissipate at least 84 kW (based on design rated voltage of 480V) when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 16,450 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 16,450 cfm ± 10%.

3/4.9.13 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 Fuel assemblies stored in the Spent Fuel Pool shall be placed in the spent fuel storage racks according to the criteria shown in Specification 5.6.1.3.

APPLICABILITY: Whenever fuel is in the Spent Fuel Pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all other fuel movement within the Spent Fuel Pool and immediately move the non-complying fuel assemblies to allowable locations in the Spent Fuel Pool in accordance with Specification 5.6.1.3.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13.1 Prior to fuel assembly movement into or within the Spent Fuel Pool, verify by administrative means that the requirements of Specification 5.6.1.3 are satisfied.

3/4.9.14 (This Specification Number is not used)

3/4.9.15 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.15 The boron concentration in the Spent Fuel Pool shall be greater than or equal to 2000 ppm.

APPLICABILITY: Whenever fuel is in the Spent Fuel Pool

ACTION:

a. With boron concentration in the Spent Fuel Pool less than 2000 ppm, immediately suspend movement of fuel in the Spent Fuel Pool and immediately initiate action to restore boron concentration to 2000 ppm or greater.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.15.1 The boron concentration of the Spent Fuel Pool shall be verified to be 2000 ppm or greater at least once per 7 days.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The requirements of Specifications 4.2.2.2, 4.2.2.3, and 4.2.3.2 shall be performed in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate Range and Power Range* channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

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

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Verify each OPERABLE Intermediate Range and Power Range* channel has been subjected to an ANALOG CHANNEL OPERATIONAL TEST per Specification Table 4.3-1 prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

*Power Range Low Setpoint only.

3/4.10.4 (THIS SPECIFICATION NUMBER IS NOT USED)

SPECIAL TEST EXCEPTIONS

3/4.10.5 THIS SPECIFICATION NUMBER IS NOT USED

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 (THIS SPECIFICATION NUMBER IS NOT USED)

LIMITING CONDITION FOR OPERATION

3.11.1.2 (THIS SPECIFICATION NUMBER IS NOT USED)

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 (THIS SPECIFICATION NUMBER IS NOT USED)

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each temporary unprotected outdoor tank shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any temporary unprotected outdoor tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.8.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each temporary unprotected outdoor tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 (THIS SPECIFICATION IS NOT USED)

LIMITING CONDITION FOR OPERATION

3.11.2.2 (THIS SPECIFICATION NUMBER IS NOT USED)

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 (THIS SPECIFICATION NUMBER IS NOT USED)

LIMITING CONDITION FOR OPERATION

3.11.2.4 (THIS SPECIFICATION NUMBER IS NOT USED)

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE - SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the GASEOUS RADWASTE TREATMENT SYSTEM shall be limited to less than or equal to 2% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the GASEOUS RADWASTE TREATMENT SYSTEM greater than 2% by volume, reduce the oxygen concentration to the above limit within 48 hours unless the hydrogen concentration is verified to be less than 4% by volume.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the GASEOUS RADWASTE TREATMENT SYSTEM shall be determined to be within the above limit by continuously monitoring the waste gases in the GASEOUS RADWASTE TREATMENT SYSTEM with the hydrogen or oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

3.11.3 (THIS SPECIFICATION NUMBER IS NOT USED)

SURVEILLANCE REQUIREMENTS

4.11.3 (Continued) (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 (THIS SPECIFICATION NUMBER IS NOT USED)

LIMITING CONDITION FOR OPERATION

3.12.1 (Continued) (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 (THIS SPECIFICATION NUMBER IS NOT USED)

SURVEILLANCE REQUIREMENTS

4.12.2 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 (THIS SPECIFICATION NUMBER IS NOT USED)

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be shown in Figures 5.1-1 and 5.1-3, respectively.

The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment building is a steel-lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 219 feet.

c. Minimum thickness of concrete walls = 4 feet 6 inches.

d. Minimum thickness of concrete dome = 3 feet 6 inches.

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner = 1/4, 3/8, and 1/2 inch for the floor, wall, and dome, respectively.

g. Net free volume = 2.704 X 10^6 cubic feet.

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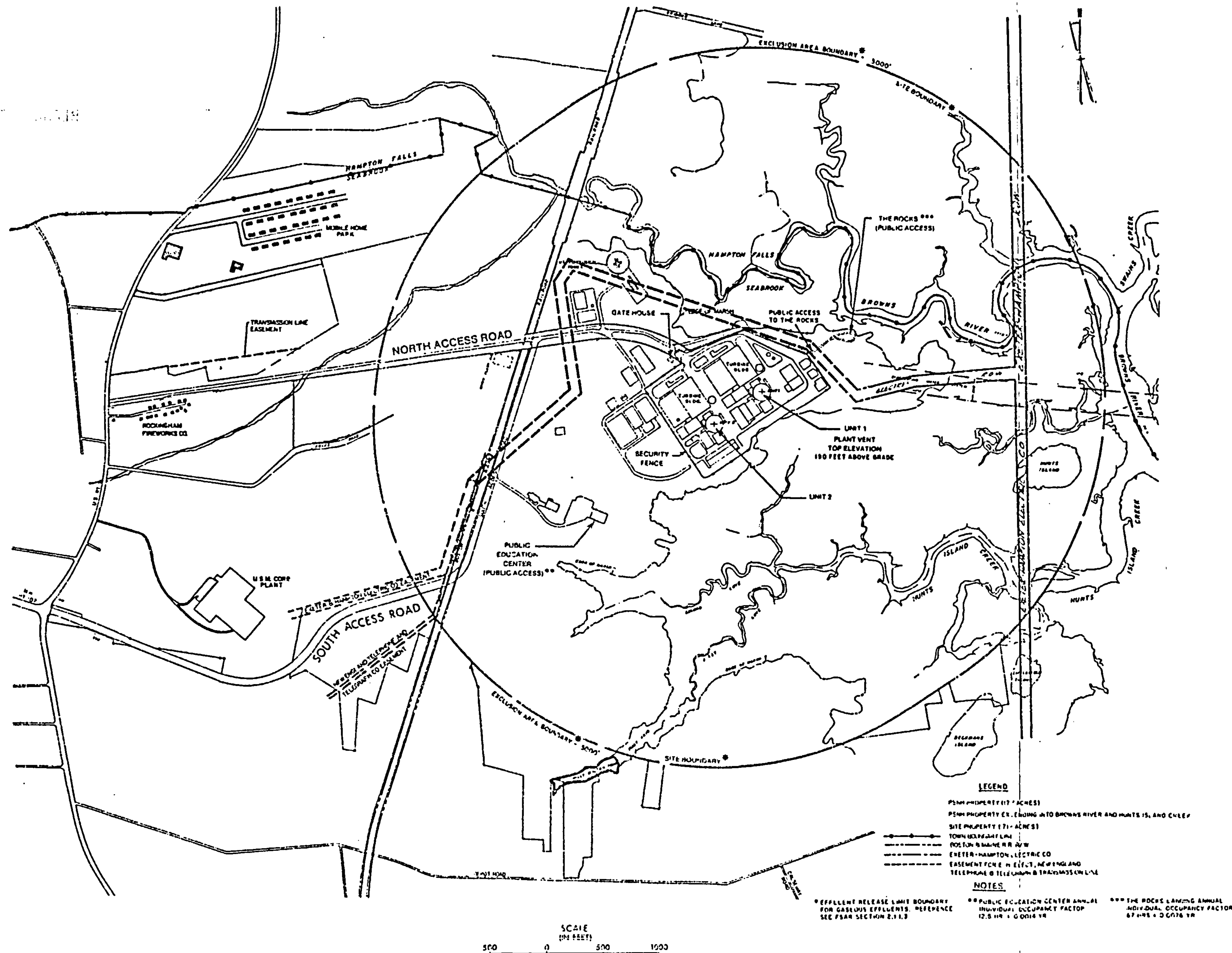

FIGURE 5.1-1

SITE AND EXCLUSION AREA BOUNDARY

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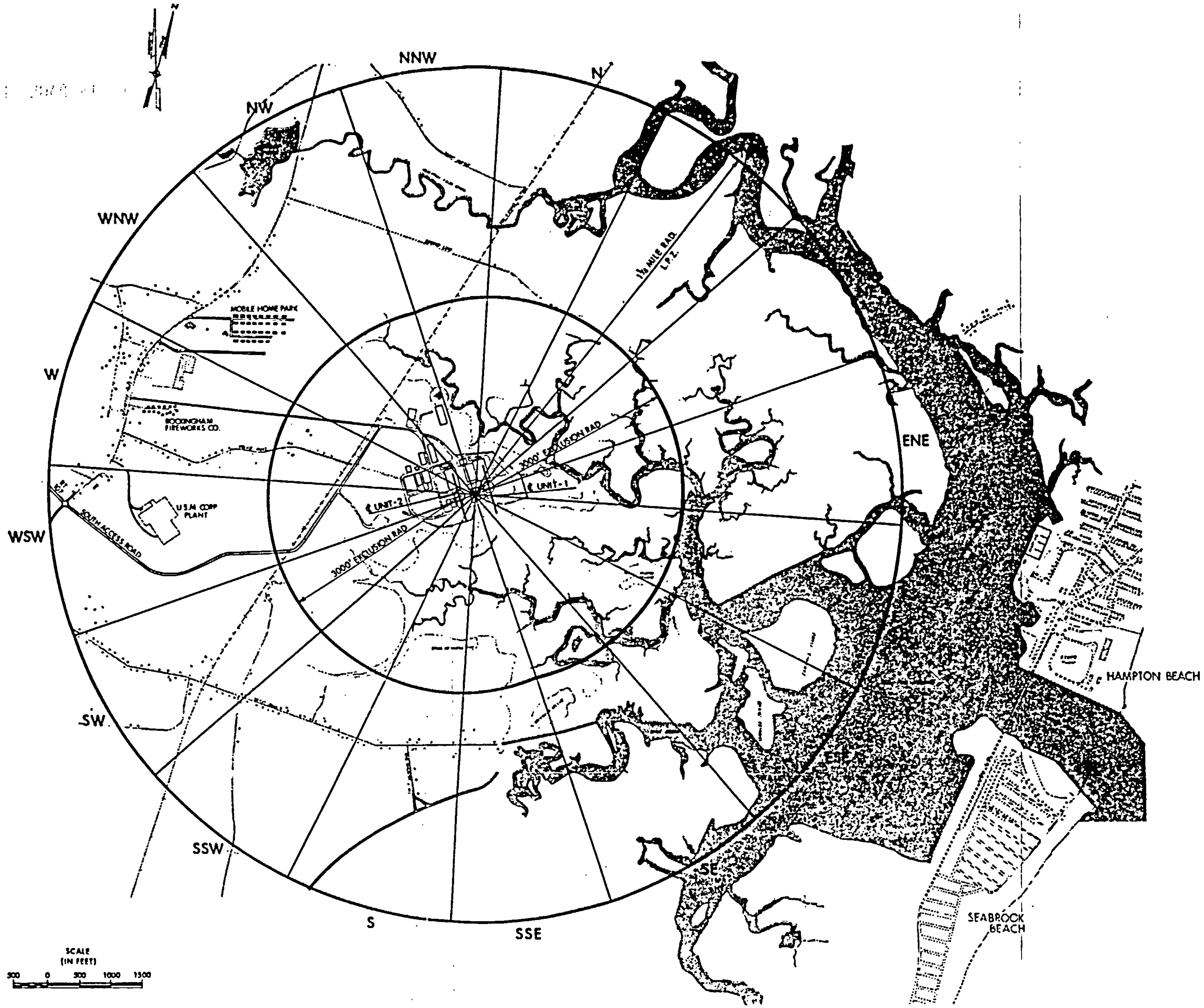

FIGURE 5.1-2

LOW POPULATION ZONE

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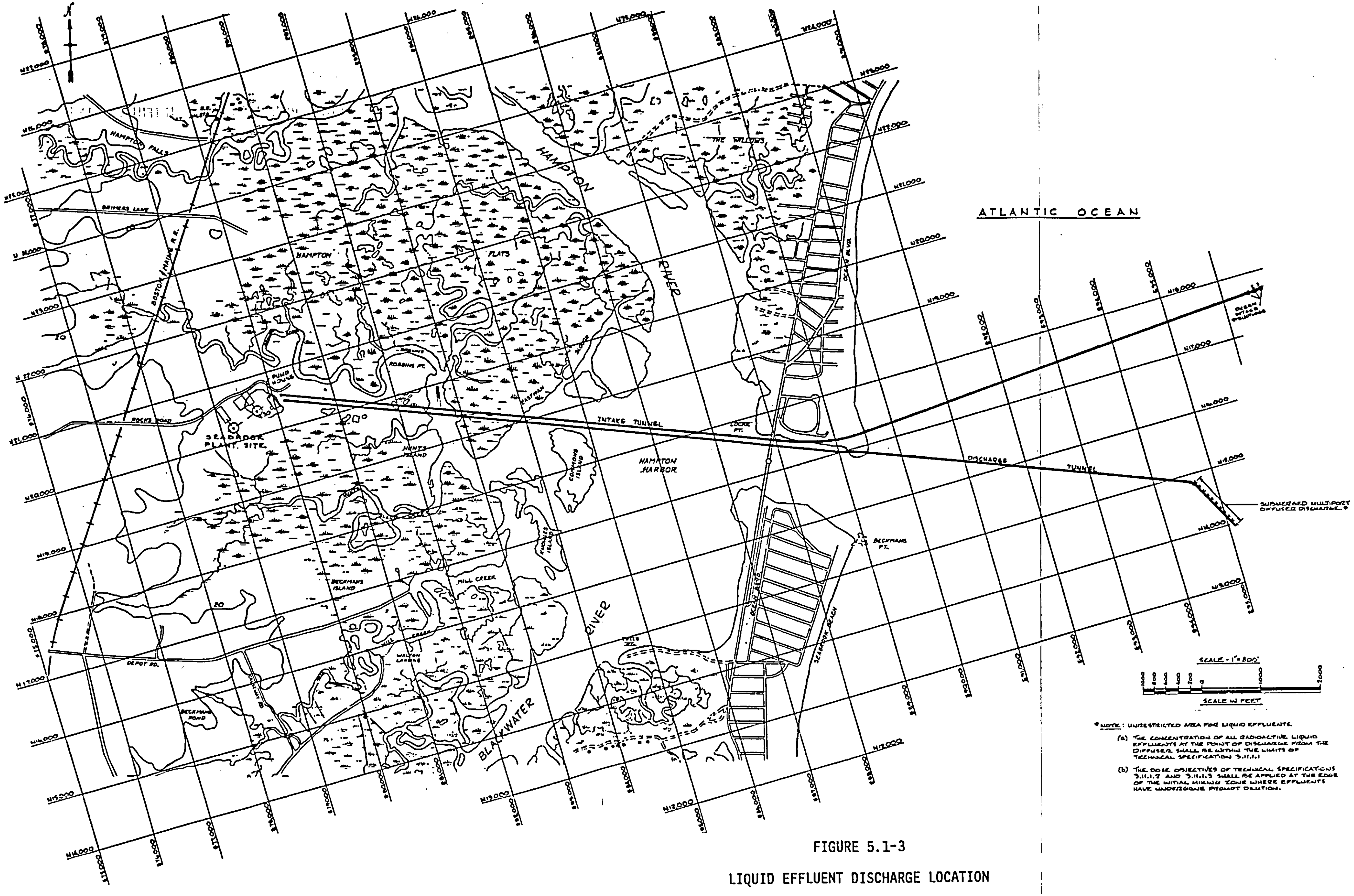

FIGURE 5.1-3

LIQUID EFFLUENT DISCHARGE LOCATION

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DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 52.0 psig and a temperature of 296°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of cylindrical Zircaloy-4, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 57 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,255 cubic feet at a nominal T_{avg} of 588.5°F.

5.5 (THIS SPECIFICATION NUMBER IS NOT USED)

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} less than 1.0 when flooded with unborated water, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

b. A k_{eff} less than or equal to 0.95 when flooded with water borated to 500 ppm, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

c. A nominal 10.35 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

b. A k_{eff} equivalent to less than or equal to 0.98 if moderated by aqueous foam, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

c. At least a nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks with a nominal 33 inches center-to-center distance (east to west) between fuel assemblies in the center column and adjacent columns.

5.6.1.3 Fresh or irradiated fuel assemblies shall be stored in the spent fuel pool in compliance with the following:

a. Any 2x2 array of Region 1 storage cells containing fuel shall comply with the storage pattern in Figure 5.6-1 and the requirements of Table 5.6-1. The reactivity ranks of fuel assemblies in the 2x2 array (rank determined using Table 5.6-1) shall be equal to or less than defined for the 2x2 array.

b. Any 2x2 array of Region 2 storage cells containing fuel shall comply with the storage requirements defined in Figure 5.6-2 and the requirements of Table 5.6-1 or with the allowable exception of evaluated assemblies stored on the periphery of Region 2 as defined in 5.6.1.3.c. The evaluated assemblies are listed in Table 5.6-2.

FUEL STORAGE

CRITICALITY

5.6.1.3 (Continued)

c. 2x2 arrays fully within the first two rows closest to the West Wall composed only of fuel assemblies documented in Table 5.6-2 or empty locations are allowed without having to meet the storage requirements defined in Figure 5.6-2 and the requirements of Table 5.6-1.

d. In addition to meeting the requirements defined in 5.6.1.3.a, fuel assemblies placed in Region 1 in the row adjacent to Region 2 shall continue the Region 2 patterns as defined in Figure 5.6-2 and shall meet the associated Region 2 reactivity class requirements.

e. Any fuel assembly (with or without an RCCA) may be replaced by an empty water cell, non-fuel hardware or a fuel rod storage basket.

<u>TABLE 5.6-1</u>

BURNUP REQUIREMENTS FOR EACH REACTIVITY CLASS
Bounding Polynomial Fits for Minimum Burnup Requirements
See Notes 1, 2 and 3 for use of Table 5.6-1

Reactivity Class [1]	Cooling Time	Coefficient A[2]		Coefficient B	Coefficient C		
RC 1[3]	N/A	N/A		N/A	N/A		
RC 2	N/A	-23.9486		7.4857	0.0000		
		Enrichment <3.6 w/o Coefficients			Enrichment ≥ 3.6 w/o Coefficients		
		A	B	C	A	B	C
RC 3	0 years	-46.4893	24.2342	-1.4689	-46.9639	23.9883	-1.4535
	2.5 years	-45.3671	23.6083	-1.4430	-44.6422	22.7925	-1.3592
	5 years	-43.3626	22.3467	-1.2912	-42.8691	21.5892	-1.2031
	10 years	-41.2729	21.3176	-1.2238	-40.4786	20.4229	-1.1214
	15 years	-37.5450	19.2208	-0.9792	-36.5543	18.2164	-0.8607
	20 years	-37.1511	19.1067	-0.9965	-35.8945	17.9317	-0.8518
RC 4	0 years	-39.4986	24.8329	-1.4714	-35.5129	22.5425	-1.2508
	2.5 years	-42.0614	26.2021	-1.7536	-31.0986	20.3032	-1.0635
	5 years	-43.5036	26.7220	-1.8423	-28.4171	18.8863	-0.9270
	10 years	-40.2450	24.8908	-1.6792	-32.5900	20.6289	-1.1778
	15 years	-39.4193	24.3389	-1.6482	-35.7271	22.0541	-1.3825
	20 years	-38.0193	23.4289	-1.5482	-33.6429	20.7970	-1.2397
RC 5	0 years	-18.6729	17.1776	-0.3238	15.4943	0.4484	1.5317
	2.5 years	-22.0079	18.6718	-0.6196	27.0014	-5.0979	2.0587
	5 years	-24.5664	19.9913	-0.8744	20.9571	-2.1108	1.6159
	10 years	-25.9493	20.7089	-1.0982	-0.9900	8.4067	0.2667
	15 years	-26.8021	21.1165	-1.2220	-13.6314	14.4202	-0.5032
	20 years	-26.3500	20.8067	-1.2333	-20.7757	17.7162	-0.9238

TABLE 5.6-1 (Continued)

BURNUP REQUIREMENTS FOR EACH REACTIVITY CLASS
Bounding Polynomial Fits for Minimum Burnup Requirements
See Notes 1, 2 and 3 for use of Table 5.6-1

Notes

1. Reactivity Classes are presented from High to Low, i.e., RC 1 is most reactive fuel, RC 5 is least reactive fuel.

2. The specific minimum burnup (Bu) required for each fuel assembly for Reactivity Classes 2-5 is calculated from the following equation:

   $$Bu = A + B \times En + C \times En^2$$

   where the coefficients A, B and C are defined above for each Reactivity Class and cooling time (if applicable) and En is defined as the nominal initial central zone enrichment. Actual cooling time is rounded down to the nearest value, e.g., an assembly with an actual cooling time of 12 years would utilize the 10 year coefficients. No uncertainties should be applied when determining the minimum burnup requirement; all appropriate uncertainties have been included during the coefficient generation.

3. Fresh or irradiated fuel with an initial enrichment of $\leq$ 5.0 w/o U-235.

TABLE 5.6-2

EVALUATED ASSEMBLIES ON PERIPHERY OF REGION 2

C01	C17	C33	C49
C02	C18	C34	C50
C03	C19	C36	C51
C04	C20	C37	C52
C05	C21	C38	C53
C06	C22	C39	C55
C07	C23	C40	C56
C09	C24	C41	C57
C10	C26	C42	C58
C11	C27	C43	C59
C12	C28	C44	C60
C13	C29	C45	C61
C14	C30	C46	C62
C15	C31	C47	C63
C16	C32	C48	C64

ALLOWABLE STORAGE PATTERN REGION 1
(See Notes 1 and 2)

Pattern "A"
See Definition 1

RC 1	RC 2
RC 2	RC 1

DEFINITIONS:

1. Allowable pattern is fuel assemblies that meet Reactivity Class (RC) 1, 2, 3, 4, or 5 checkerboarded with fuel assemblies that meet RC 2, 3, 4, or 5. Requirements for all RC are defined in Table 5.6-1. Diagram is for illustration only.

NOTES

1. There are no interface limitations within Region 1 between rack modules or within racks. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. Replacement of any fuel assembly by an empty water hole, non-fuel hardware or fuel rod storage basket is acceptable.

FIGURE 5.6-1

ALLOWABLE STORAGE PATTERNS REGION 2 (See Notes 1, 2)

Pattern "B"

RC 3 RCCA	RC 3 RCCA
RC 3	RC 3

See Definition 1

Pattern "C"

RC 4	RC 4
RC 4	RC 4 RCCA

See Definition 2

Pattern "D"

RC 5	RC 5
RC 5	RC 5

See Definition 3

DEFINITIONS

1. Allowable pattern is fuel assemblies that meet Reactivity Class (RC) 3, 4, or 5 in each of the 2x2 array locations combined with two RCCAs placed in any two locations within the 2x2 array. Requirements for all RC are defined in Table 5.6-1. Replacement of any fuel assembly (with or without an RCCA) by an empty water hole, non-fuel hardware or fuel rod storage basket is acceptable. Diagram is for illustration only.

2. Allowable pattern is fuel assemblies that meet Reactivity Class (RC) 4 or 5 in each of the 2x2 array locations with one RCCA placed anywhere in the 2x2 array. Requirements for all RC are defined in Table 5.6-1. Replacement of any fuel assembly (with or without an RCCA) by an empty water hole, non-fuel hardware or fuel rod storage basket is acceptable. Diagram is for illustration only.

3. Allowable pattern is Reactivity Class (RC) 5 in each of the 2x2 array locations. Minimum burnup for RC 5 is defined in Table 5.6-1 as a function of nominal initial central zone enrichment and cooling time. Replacement of any fuel assembly by an empty water hole, non-fuel hardware or fuel rod storage basket is acceptable. Diagram is for illustration only.

NOTES

1. The storage arrangements of fuel within a rack module may contain more than one pattern. There are no interface limitations within Region 2 between rack modules or within racks. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

2. All permanent and transient configurations for fuel placed within Region 2 must meet the requirements of Figure 5.6-2 and Table 5.6-1.

FIGURE 5.6-2

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 14 feet 6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1236 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	200 leak tests.	Pressurized to ≥ 2250 psig.
	10 hydrostatic pressure tests.	Pressurized to ≥ 3106 psig.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to ≥ 1481 psig.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall station operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions for departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR and updated in accordance with the requirements of 10 CFR 50.71.

b. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 STATION STAFF

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a and 6.2.2.d for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. An individual (Shift Technical Advisor (STA)) shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. The STA position shall be manned in MODES 1, 2, 3, and 4 unless the SM or the individual with a Senior Operator license meets the qualifications for the STA.

e. While the unit is in MODE 1, 2, 3 or 4, a licensed senior operator, either the SM or SRO, shall be on shift having had at least 6 months of hot operating experience.

f. The Operations Manager shall meet one of the following:

   1. Hold a senior operator license,
   2. Have held a senior operator license on a similar unit (PWR), or
   3. Have been certified for equivalent senior operator knowledge.

g. The Assistant Operations Manager shall hold a senior reactor operator license.

TABLE 6.2-1

DELETED

6.2.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.2.4 (THIS SPECIFICATION NUMBER IS NOT USED)

6.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.4 (THIS SPECIFICATION NUMBER IS NOT USED)

6.5 (THIS SPECIFICATION NUMBER IS NOT USED)

6.6 (THIS SPECIFICATION NUMBER IS NOT USED)

6.7 PROCEDURES AND PROGRAMS

6.7.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Not used;

d. Not used;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Assurance Program for effluent and environmental monitoring;

h. Fire Protection Program implementation; and

i. Technical Specification Improvement Program implementation.

6.7.2 (THIS SPECIFICATION NUMBER IS NOT USED)

6.7.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.7.4 (THIS SPECIFICATION NUMBER IS NOT USED)

6.7.5 (THIS SPECIFICATION NUMBER IS NOT USED)

PROCEDURES AND PROGRAMS

6.7.6 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the RHR and containment spray, Safety Injection, chemical and volume control. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program that will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

5) Procedures defining corrective actions for all off-control point chemistry conditions, and

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

d. Backup Method for Determining Subcooling Margin

A program that will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1) Training of personnel, and

2) Procedures for monitoring.

e. (Not Used)

f. Accident Monitoring Instrumentation

A program which will ensure the capability to monitor plant variables and systems operating status during and following an accident. This program shall include those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials (Category 2 and 3 instrumentation as defined in Regulatory Guide 1.97, Revision 3)* and provide the following:

1) Preventive maintenance and periodic surveillance of instrumentation,

2) Preplanned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

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*Seabrook has taken exception to the categorization of instrumentation provided in Regulatory Guide 1.97, Revision 3. The Seabrook exceptions are provided in FSAR Table 7.5-1, which has been reviewed by the NRC staff in SER Supplement No. 5.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

g. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to ten times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001-20.2402,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from the unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) 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 ODCM at least every 31 days. Determination of projected dose contribution from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY shall be limited to the following:

a) For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to 1500 mrems/yr to any organ,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) (Not Used), and

11) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

h. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

i. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements and acceptance criteria, using methodologies described in applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or 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*;

b. Within 31 days following addition of the new fuel oil to the storage tank(s), 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 stored fuel oil is < 10 mg/l when tested every 31 days using methodologies described in applicable ASTM Standards.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

* For fuel oil that has been dyed, the centrifuge method for quantifying water and sediment in distillate fuels specified in the applicable ASTM Standard is an acceptable method of performing this verification.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

j. Technical Specification (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR (UFSAR) 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 UFSAR.

d. Proposed changes that meet the criteria of Specification 6.7.6j.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).

k. Steam Generator (SG) Program

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total or 500 gpd through any one SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2, “Reactor Coolant System Operational Leakage.”

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:

Tubes with service-induced flaws located greater than 15.21 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.21 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet except for any portions of the tube that are exempt from inspection by alternate repair criteria, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 54 effective full power months, which defines the inspection period. If none of the SG tubes have ever experienced cracking other than in regions that are exempt from inspection by alternate repair criteria and the SG inspection was performed with enhanced probes, the inspection period may be extended to 72 effective full power months. Enhanced probes have a capability to detect flaws of any type equivalent to or better than array probe technology. The enhanced probes shall be used from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

weld at the tube outlet except any portions of the tube that are exempt from inspection by alternate repair criteria. If there are regions where enhanced probes cannot be used, the tube inspection techniques shall be capable of detecting all forms of existing and potential degradation in that region.

3. If crack indications are found in portions of the SG tube excluding any region that is exempt from inspection by alternate repair criteria, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage, but may be deferred to the following refueling outage if the 100% inspection of all SGs was performed with enhanced probes as described in paragraph d.2. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary leakage.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

l. 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 Makeup Air and Filtration System (CREMAFS), 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 in-leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREMAFS, operating at a flow rate of less than or equal to 600 CFM at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 36 month assessment of the CRE boundary.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

e. The quantitative limits on unfiltered air in-leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air in-leakage measured by the testing described in paragraph c. The unfiltered air in-leakage limit for radiological challenges is the in-leakage 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 SR 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered in-leakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

m. Reactor Coolant Pump Flywheel Inspection Program

In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 at least once every 20 years. In lieu of Position C.4.b(1) and C.4.b(2), this inspection shall be by either of the following examinations:

a. An in-place examination, utilizing ultrasonic testing, over the volume from the inner bore of the flywheel to the circle of one-half the outer radius; or

b. A surface examination, utilizing magnetic particle testing and/or penetrant testing, of the exposed surfaces of the disassembled flywheel.

n. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

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.

o. Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system ACTIONS. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

c. Provisions to ensure that an inoperable supported system's allowed outage time is not inappropriately extended as a result of multiple support system inoperabilities, and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

b. A required system redundant to the system(s) in turn supported by the inoperable support system is also inoperable, or

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate ACTIONS of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ACTIONS to enter are those of the support system.

6.8 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.8.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.8.1.1 A summary report of station 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 station.

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.8.1.2 Annual Reports covering the activities of the station as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

Reports required on an annual basis shall include:

The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (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 limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radio-iodine isotope concentration (μCi/gm) 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.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.8.1.3 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 analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.8.1.4 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

6.8.1.5 Deleted

CORE OPERATING LIMITS REPORT

6.8.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

1. Cycle dependent Overpower ΔT and Overtemperature ΔT trip setpoint parameters and function modifiers for operation with skewed axial power profiles for Table 2.2-1 of Specification 2.2.1.

2. Cycle dependent maximum allowable combination of thermal power, pressurizer pressure and the highest operating loop average temperature (T_{avg}) for Specification 2.1.1.

3. SHUTDOWN MARGIN and minimum boron concentration limits for MODES 1, 2, 3, and 4 for Specification 3.1.1.1.

4. SHUTDOWN MARGIN and minimum boron concentration limits for MODE 5 for Specification 3.1.1.2.

5. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3.1.1.3.

6. The minimum boron concentration for Modes 4, 5, and 6 for Specification 3.1.2.7.

7. Shutdown Rod Insertion limit for Specification 3.1.3.5.

8. Control Rod Bank Insertion limits for Specification 3.1.3.6.

9. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1

10. Heat Flux Hot Channel Factor, F_Q^{RTP} and K(Z) for Specification 3.2.2.

11. Nuclear Enthalpy Rise Hot Channel Factor, and $F_{\Delta H}^{RTP}$ for Specification 3.2.3.

12. Cycle dependent DNB-related parameters for reactor coolant system average temperature (T_{avg}), and pressurizer pressure for Specification 3.2.5.

13. The boron concentration limits for MODES 1, 2 and 3 for Specification 3.5.1.1.

14. The boron concentration limits for MODES 1, 2, 3 and 4 for Specification 3.5.4.

15. The boron concentration limits for MODE 6 for Specification 3.9.1.

6.8.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-12945P-A, "Code Qualification Document for Best Estimate LOCA Analysis," Volume 1, Revision 2, and Volumes 2 through 5, Revision 1; Bajorek, S. M., et al, 1998.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

2. WCAP-10079-P-A, (Proprietary) and WCAP-10080-A (Nonproprietary), "NOTRUMP: A Nodal Transient Small Break and General Network Code", August 1985.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

3. YAEC-1363-A, "CASMO-3G Validation," April, 1988.

   YAEC-1659-A, "SIMULATE-3 Validation and Verification," September, 1988.

   WCAP-11596-P-A, (Proprietary), "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores", June, 1988.

   WCAP-10965-P-A, (Proprietary), "ANC: A Westinghouse Advanced Nodal Computer Code", September, 1986.

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1,2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Bank Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

4. Seabrook Station Updated Final Safety Analysis Report, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant System".

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5

6.8.1.6.b (Continued)

5. YAEC-1241, "Thermal-Hydraulic Analysis of PWR Fuel Elements Using the CHIC-KIN Code", R. E. Helfrich, March, 1981.

   WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis", October, 1999.

   WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

   Methodology for Specification:
   2.1 - Safety Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor
   3.2.5 - DNB Parameters

6. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 For PWR Applications," October, 1992.

   WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure", April, 1989.

   WCAP-8745-P-A, Design Basis for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

7. YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station," October, 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.3.5 - Shutdown Bank Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b (Continued)

8. YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications," December, 1992.

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Bank Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

9. YAEC-1752, "STAR Methodology Application for PWRs, Control Rod Ejection, Main Steam Line Break," October, 1990.

   Methodology for Specification:
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Bank Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

10. YAEC-1855PA, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis," October, 1992.

    ANP-3243P, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis Supplement to YAEC-1855PA," Revision 1, May 2014.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

11. YAEC-1624P, "Maine Yankee RPS Setpoint Methodology Using Statistical Combination of Uncertainties - Volume 1 - Prevention of Fuel Centerline Melt," March, 1988.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b (Continued)

12. NYN-95048, Letter from T. C. Feigenbaum (NAESCo) to NRC, "License Amendment Request 95-05: Positive Moderator Temperature Coefficient", May 30, 1995.

    Methodology for Specification:
    3.1.1.3 - Moderator Temperature Coefficient

13. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report". April, 1995, (Westinghouse Proprietary).

    WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™", July 2006.

    Methodology for Specification:
    3.2.2 - Heat Flux Hot Channel Factor

14. WCAP-10216-P-A, Revision 1A (Proprietary), "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification", February, 1994.

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor

15. WCAP-9272-P-A, (Proprietary), "Westinghouse Reload Safety Evaluation Methodology", July, 1985.

    Methodology for Specifications:
    2.1 - Safety Limits
    3.1.1.1 - SHUTDOWN MARGIN for MODES 1,2,3, and 4
    3.1.1.2 - SHUTDOWN MARGIN for MODE 5
    3.1.1.3 - Moderator Temperature Coefficient
    3.1.2.7 - Isolation of Unborated Water Sources - Shutdown
    3.1.3.5 - Shutdown Bank Insertion Limit
    3.1.3.6 - Control Rod Insertion Limits
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor
    3.2.5 - DNB Parameters
    3.5.1.1 - Accumulators for MODES 1, 2, and 3
    3.5.4 - Refueling Water Storage Tank for MODES 1, 2, 3, and 4
    3.9.1 - Boron Concentration

16. WCAP-13749-P-A, (Proprietary) "Safety Evaluation Supporting the Conditional Exemption of the Most Negative Moderator Temperature Coefficient Measurement," March, 1997.

    Methodology for Specifications:
    3.1.1.3 - Moderator Temperature Coefficient

6.8.1.6.b (Continued)

17. License Amendment 169 issued 09/22/21 (ADAMS Accession No. ML21190A177)

    Methodology for Specification:

    3.2.2 - Heat Flux Hot Channel Factor

6.8.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

6.8.1.7 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.7.6.k, “Steam Generator (SG) Program.” The report shall include:

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

4. The number of tubes plugged during the inspection outage.

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG;

6.8.1.7 (Continued)

f. The results of any SG secondary side inspections;

g. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report;

h. The calculated accident induced leakage rate from the portion of the tubes below 15.21 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.49 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined; and

i. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator within the time period specified for each report.

6.9 (THIS SPECIFICATION NUMBER IS NOT USED)

6.10 RADIATION PROTECTION PROGRAM

6.10.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.11 HIGH RADIATION AREA

6.11.1 Pursuant to paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) and (b), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less-than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface that the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit.

6.11.2 In addition to the requirements of Specification 6.11.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface that the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

HIGH RADIATION AREA

6.11.2 (Continued)

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.12 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Assurance Program (OQAP). This documentation shall contain:

    1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

    2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the Onsite Review Group and approval of the plant manager.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Control Program (OQAP). This documentation shall contain:

    1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

    2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the Onsite Review Group and the approval of the plant manager.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and each affected page shall indicate the revision number the change was implemented.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Onsite Review Group. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

*Licensees may choose to submit the information called for in this Specification as part of the FSAR update, pursuant to 10 CFR 50.71.

6.14.1 (Continued)

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the Onsite Review Group.

b. Shall become effective upon review and acceptance by the Onsite Review Group.

6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

CONTAINMENT LEAKAGE RATE TESTING PROGRAM

6.15 (Continued)

Overall air lock leakage rate acceptance criterion is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

Each containment 8-inch purge supply and exhaust isolation valve leakage rate acceptance criterion is $\leq 0.01\ L_a$ when tested at P_a.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. NPF-86

SEABROOK STATION, UNIT 1

NEXTERA ENERGY SEABROOK, LLC

DOCKET NO. 50-443

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

SEABROOK STATION, UNIT NO. 1

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

TABLE OF CONTENTS

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of non-radiological environmental values during operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement - Operating Licensing Stage (FES-OL) and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters are regulated by way of the licensee's NPDES permit.

2.0 Environmental Protection Issues

In the FES-OL (NUREG-0895) dated December, 1982, the staff considered the environmental impacts associated with the operation of Seabrook Station, Unit No. 1. No aquatic/water quality, terrestrial, or noise issues were identified.

Aquatic matters are addressed by the effluent limitations and monitoring requirements contained in NPDES Permit No. NH0020338 issued by the U. S. Environmental Protection Agency (Region I) as amended. The NRC will rely on the U.S.E.P.A and the NPDES Permit for regulation of matters involving water quality and aquatic biota.

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP*. Changes in station design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

Before engaging in additional construction or operational activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measurable nonradiological environmental effects are confined to the onsite areas previously disturbed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a

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* This provision does not relieve the licensee of the requirements of 10 CFR 50.59.

written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level; or (3) a matter, not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of the Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

3.2 Reporting Related to the NPDES Permit and State Certification

Changes to, or renewals of, the NPDES Permits or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The licensee shall notify the NRC of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES Permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, and local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; fish kills; increase in nuisance organisms or conditions; and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decisions made by the U. S. Environmental Protection Agency and the State of New Hampshire under the authority of the Clean Water Act, for any requirements for aquatic monitoring.

4.2.2 Terrestrial Monitoring

Terrestrial monitoring is not required.

4.2.3 Noise Monitoring

Noise monitiorinq is not required

5.0 Administrative Procedures

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the EPP. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the station. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Requests for changes in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The period of the first report shall begin with the date of issuance of the operating license, and the initial report shall be submitted prior to May 1 of the year following issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 (if any) of this EPP for the report period, including a comparison with related preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends toward irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of mitigating action.

The Annual Environmental Operating Report shall also include:

(1) A list of EPP noncompliances and the corrective actions taken to remedy them.

(2) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental question.

(3) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing results shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall: (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact, and plant operating characteristics; (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event; (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems; and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this Subsection. The NRC shall be provided with a copy of such report at the same time it is submitted to the other agency.

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. NPF-86

NextEra Energy Seabrook, LLC, shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
50	NAESCO is authorized to relocate certain technical specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated October 17, 1996, and evaluated in the staff's Safety Evaluation attached to this amendment.	The amendment shall be implemented within 60 days from March 12, 1997
112	NextEra Energy Seabrook, LLC shall maintain the operational limit of primary-to-secondary leakage at 150 gallons per day per Steam Generator and if this limit is exceeded, NextEra Energy Seabrook, LLC will take the appropriate actions in accordance with TS 3.4.6.2, "Reactor Coolant System Leakage."	This amendment shall be implemented within 90 days from September 29, 2006
119	Upon implementation of Amendment No. 119 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 4.7.6.1.g, in accordance with TS 6.7.6.l.c. (i), the assessment of CRE habitability as required by Specification 6.7.6.lc. (ii), and the measurement of CRE pressure as required by Specification 6.7.6.l.d, shall be considered met. Following implementation: (a) The first performance of SR 4.7.6.1.g, in accordance with Specification 6.7.6.l.c. (i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from August 2003, the date of the most recent successful tracer gas test, as stated in the December 9, 2003 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.	This amendment shall be implemented within 6 months from July 30, 2008

Amendment Number	Additional Condition	Implementation Date
	(continued) (b) The first performance of the periodic assessment of CRE habitability, Specification 6.7.6.I.c. (ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from August 2003, the date of the most recent successful tracer gas test, as stated in the December 9, 2003 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years. (c) The first performance of the periodic measurement of CRE pressure, Specification 6.7.6.I.d, shall be within 18 months, plus the 138 days allowed by SR 4.0.2, as measured from August 2003, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.	
159	The licensee will perform the following actions to confirm the continued applicability of the MPR/FSEL large-scale testing program conclusions to Seabrook structures (i.e., that future expansion behavior of ASR-affected concrete structures at Seabrook aligns with observations from the MPR/FSEL large-scale testing program and that the associated expansion limits remain applicable). The licensee shall notify the NRC each time an assessment or corroboration action is completed. (a) Conduct assessments of expansion behavior using the approach provided in Appendix B of Report MPR-4273, Revision 1 (Seabrook FP#101050), to confirm that future expansion behavior of ASR-affected structures at Seabrook Station is comparable to what was observed in the MPR/FSEL large-scale testing program and to check margin for future expansion. Seabrook completed the first expansion assessment in March 2018; and will complete subsequent expansion assessments every ten years thereafter. (b) Corroborate the concrete modulus-expansion correlation used to calculate pre-instrument through-thickness expansion, as discussed in Report MPR-4153, Revision 3 (Seabrook FP#100918). The corroboration will cover at least 20 percent of extensometer locations on ASR-affected structures and will use the approach provided in Appendix C of Report MPR-4273, Revision 1 (Seabrook FP#101050). Seabrook will complete the initial study no later than 2025 and a follow-up study 10 years thereafter.	This amendment shall be implemented within 90 days of March 11, 2019

Amendment Number	Additional Condition	Implementation Date
	(continued) (c) NextEra shall undertake the monitoring required by MPR-4273, Revision 1, Appendix B, Check 3, for control extensometers every six months. (d) If stress analyses conducted pursuant to the Structural Evaluation Methodology show that the stress in the rebar from ASR-induced expansion and other loads will exceed the yield strength of the rebar, NextEra must develop a monitoring program sufficient to ensure that rebar failure or yielding does not occur, or is detected if it has already occurred, in the areas at-risk of rebar failure or yielding. (e) If the ASR expansion rate in any area of a Seabrook seismic Category I structure significantly exceeds 0.2 mm/m (0.02%) through-thickness expansion per year, NextEra's Management will perform an engineering evaluation focused on the continued suitability of the six-month monitoring interval for Tier 3 areas. If the engineering evaluation concludes that more frequent monitoring is necessary, it shall be implemented under the Structures Monitoring Program. (f) Each core extracted from Seabrook Unit 1 will be subjected to a petrographic analysis to detect internal microcracking and delamination.	