FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-67

The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-67 issued March 1, 1976, has now found that:

a. The application to renew License No. DPR-67 filed by the Florida Power and Light Company (FPL or the licensee), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter 1, and all required notifications to other agencies or bodies have been duly made;

b. 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 St. Lucie Plant Unit No.1, and that any changes made to the plant's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations;

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

d. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

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

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

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

h. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Renewed Facility Operating License No. DPR-67, subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

On the basis of the foregoing findings regarding this facility, Facility Operating License No. DPR-67, issued March 1, 1976, is superseded by Renewed Facility Operating License No. DPR-67, which is hereby issued to FPL to read as follows:

1. This renewed license applies to the St. Lucie Plant, Unit No. 1, a pressurized water nuclear reactor, and associated steam generators and electrical generating equipment (the facility). The facility is located on the licensee's site on Huchinson Island in St. Lucie County, Florida, and is described in the Updated Final Safety Analysis Report, as supplemented and amended, and the Environmental Report, as supplemented and amended.

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

   A. Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location on the St. Lucie site in accordance with the procedures and limitations set forth in this renewed license;

   B. 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;

   C. 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;

   D. 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

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

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

applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 3020 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 252, are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

Appendix B, the Environmental Protection Plan (Non-Radiological), contains environmental conditions of the renewed license. If significant detrimental effects or evidence of irreversible damage are detected by the monitoring programs required by Appendix B of this license, FPL will provide the Commission with an analysis of the problem and plan of action to be taken subject to Commission approval to eliminate or significantly reduce the detrimental effects or damage.

C. Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 28, 2003, describes certain future activities to be completed before the period of extended operation. FPL shall complete these activities no later than March 1, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on March 28, 2003, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed license. Until that update is complete, FPL may make changes to the programs described in such supplement without prior Commission approval, provided that FPL evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Sustained Core Uncovery Actions

Procedural guidance shall be in place to instruct operators to implement actions that are designed to mitigate a small-break loss-of-coolant accident prior to a calculated time of sustained core uncovery.

E. Fire Protection

Florida Power & Light Company (FPL) St. Lucie Plant Unit 1 shall implement and maintain in effect all provisions ofthe approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment requests dated March 22, 2013, and May 2, 2017, and supplements dated June 14, 2013, February 24, 2014, March 25, 2014, April 25, 2014, July 14, 2014, August 27, 2014, September 10, 2014, October 10, 2014, March 10, 2015, April 1, 2015, April 20, 2015, May 12, 2015, August 21, 2015, October 22, 2015, and as approved in the safety evaluations (SE) dated March 31, 2016, and October 23, 2017. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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

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

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

Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program.

   Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to NFPA 805, Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

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

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

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

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC SE dated March 31, 2016 to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) and (3) below, risk informed changes to Florida Power & Light Company fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2) above;

(2) The licensee shall implement the modifications to its facility, as described in Table S-1 , "Plant Modifications Committed," Attachment S, of Florida Power & Light letter L-2017-058, dated May 2, 2017, to complete the transition to full compliance with 10 CFR 50.48(c) prior to startup of SL1-28 (spring 2018) and SL2-24 (fall 2018) refueling outages after issuance of the SE. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications; and

(3) The licensee shall implement the items listed in Attachment S , Table S-2, "Implementation Items," of FPL letter L-2015-211 dated August 21, 2015, with the exception of items 18 and 20, within 12 months after NRC approval unless that falls within a scheduled outage window, then in that case, completion will occur 60 days after startup from that scheduled outage. Implementation Item 18 is an exception because it is associated with modifications in Table S-1 and will be completed in accordance with Transition License Condition 2) above. Item 20 is also an exception because it is required to be completed prior to self-approval and will be completed prior to the startup of SL2-24 (fall 2018).

F. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to 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, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Florida Power and Light & FPL Energy Seabrook Physical Security Plan, Training and Qualification Plan and Safeguards Contingency Plan - Revision 3," submitted by letter dated May 18, 2006. St. Lucie 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 St. Lucie CSP was approved by License Amendment No. 211 as supplemented by a Clarification approved by License Amendment Nos. 214 and 222.

G. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
  1. Pre-defined coordinated fire response strategy and guidance
  2. Assessment of mutual aid fire fighting assets
  3. Designated staging areas for equipment and materials
  4. Command and control
  5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
  1. Protection and use of personnel assets
  2. Communications
  3. Minimizing fire spread
  4. Procedures for implementing integrated fire response strategy
  5. Identification of readily-available pre-staged equipment

6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

H. Control Room Habitability

Upon implementation of Amendment No. 205, adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 4.7.7.1.e, in accordance with TS 6.8.4.m, the assessment of CRE habitability as required by Specification 6.8.4.m.c. (ii), and the measurement of CRE pressure as required by Specification 6.8.4.m.d, shall be considered met. Following implementation:

(a) The first performance of SR 4.7.7.1.e, in accordance with Specification 6.8.4.m.c(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from September 2003, the date of the most recent successful tracer gas test, as stated in FPL letters to NRC dated December 9, 2003, and October 29, 2004, in response to Generic Letter 2003-01.

(b) The first performance of the periodic assessment of CRE habitability, Specification 6.8.4.m.c(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from September 2003, the date of the most recent successful tracer gas test, as stated in FPL letters to NRC dated December 9, 2003, and October 29, 2004, in 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 6.8.4.c.d, shall be within 36 months in a staggered test basis, plus the 138 days allowed by SR 4.0.2, as measured from June 30, 2006, which is the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.

I. RODEX2 Safety Analyses

RODEX2 has been specifically approved for use for St. Lucie Unit 1 licensing basis analyses. Upon NRC's approval of a generic supplement to the RODEX2 code and associated methods that accounts for thermal conductivity degradation (TCD), FPL will within six months:

(a) Demonstrate that St. Lucie Unit 1 safety analyses remain conservatively bounded in licensing basis analyses when compared to the NRC-approved generic supplement to the RODEX2 methodology, or

(b) Provide a schedule for the re-analysis using the NRC-approved generic supplement to the RODEX2 methodology for any of the affected licensing basis analyses.

J. FPL is authorized to implement the Risk Informed Completion Time Program as approved in License Amendments No. 247 and 252 subject to the following conditions:

1. Deleted

2. The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant; and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval via a license amendment.

4. This renewed license is effective as of the date of issuance and shall expire at midnight on March 1, 2036.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A, Technical Specifications
2. Appendix B, Environmental Protection Plan

Date of Issuance: October 2, 2003

ST. LUCIE PLANT

UNIT 1

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

LICENSE NO. DPR-67

INDEX

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

DEFINITIONS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

INDEX

INDEX

INDEX

INDEX

DESIGN FEATURES

INDEX

ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

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

AXIAL SHAPE INDEX

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

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

AZIMUTHAL POWER TILT - T_q

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

$$\text{AZIMUTHAL POWER TILT} = \max \left| \frac{\text{Power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}} - 1 \right|$$

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

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

CONTAINMENT VESSEL INTEGRITY

1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:

a. All containment vessel 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 position except for valves that are open on an intermittent basis under administrative control.

b. All containment vessel equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include shared (4 fingered) control element assemblies (CEAs) withdrawn into the upper guide structure (UGS) or evolutions performed with the UGS in place such as CEA latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

1.9a The COLR is the unit-specific document that provides cycle specific parameter limits for the current operating reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE-133

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

ENGINEERED SAFETY FEATURES RESPONSE TIME

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

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

IDENTIFED LEAKAGE

1.15 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 system (Primary-to-secondary leakage).

INSERVICE TESTING PROGRAM

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

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

OPERABLE – OPERABILITY

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

OPERATIONAL MODE

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

PHYSICS TESTS

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

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

PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packing 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

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

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3020 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

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

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

SHIELD BUILDING INTEGRITY

1.28 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

1.30 SITE BOUNDARY means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

1.35 Unrestricted area means an area, access to which is neither limited nor controlled by the licensee.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.36 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

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
4/M*	At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year
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
P**	Completed prior to each release
SFCP	In accordance with the Surveillance Frequency Control Program
N.A.	Not applicable

* For Radioactive Effluent Sampling

** For Radioactive Batch Releases Only

TABLE 1.2

OPERATIONAL MODES

MODE		REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1.	POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 325°F
2.	STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 325°F
3.	HOT STANDBY	< 0.99	0	$\geq$ 325°F
4.	HOT SHUTDOWN	< 0.99	0	325°F > T_{avg} > 200°F
5.	COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6.	REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

* Excluding decay heat.

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

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

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

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

Amendment No. ~~115~~, 151

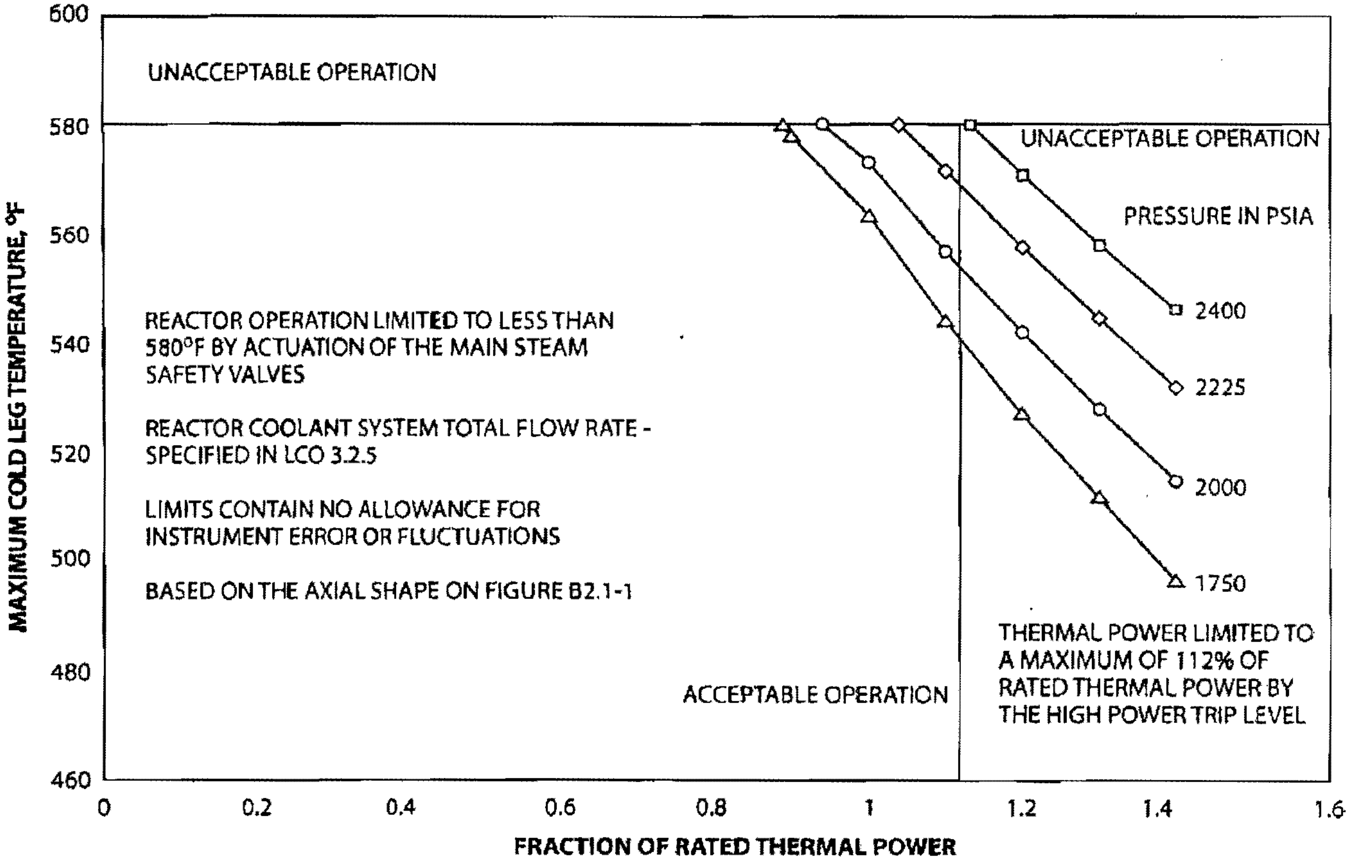


FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

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

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

ACTION:

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

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level – High (1) Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of < 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow – Low (1) Four Reactor Coolant Pumps Operating	≥ 95% of minimum reactor coolant flow with 4 pumps operating *	≥ 95% of minimum reactor coolant flow with 4 pumps operating *
4. Pressurizer Pressure – High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure – High	≤ 3.3 psig	≤ 3.3 psig
6. Steam Generator Pressure – Low (2)	≥ 600 psia	≥ 600 psia
7. Steam Generator Water Level – Low	≥ 35.0% Water Level – each steam generator	≥ 35.0% Water Level – each steam generator
8. Local Power Density – High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

* For minimum reactor coolant flow with 4 pumps operating, refer to Technical Specification LCO 3.2.5.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.
9a. Steam Generator Pressure Difference High (1) (logic in TM/LP)	$\leq$ 135 psid	$\leq$ 135 psid
10. Loss of Turbine – Hydraulic Fluid Pressure – Low (3)	$\geq$ 800 psig	$\geq$ 800 psig
11. Rate of Change of Power – High (4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq$ 1% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER.

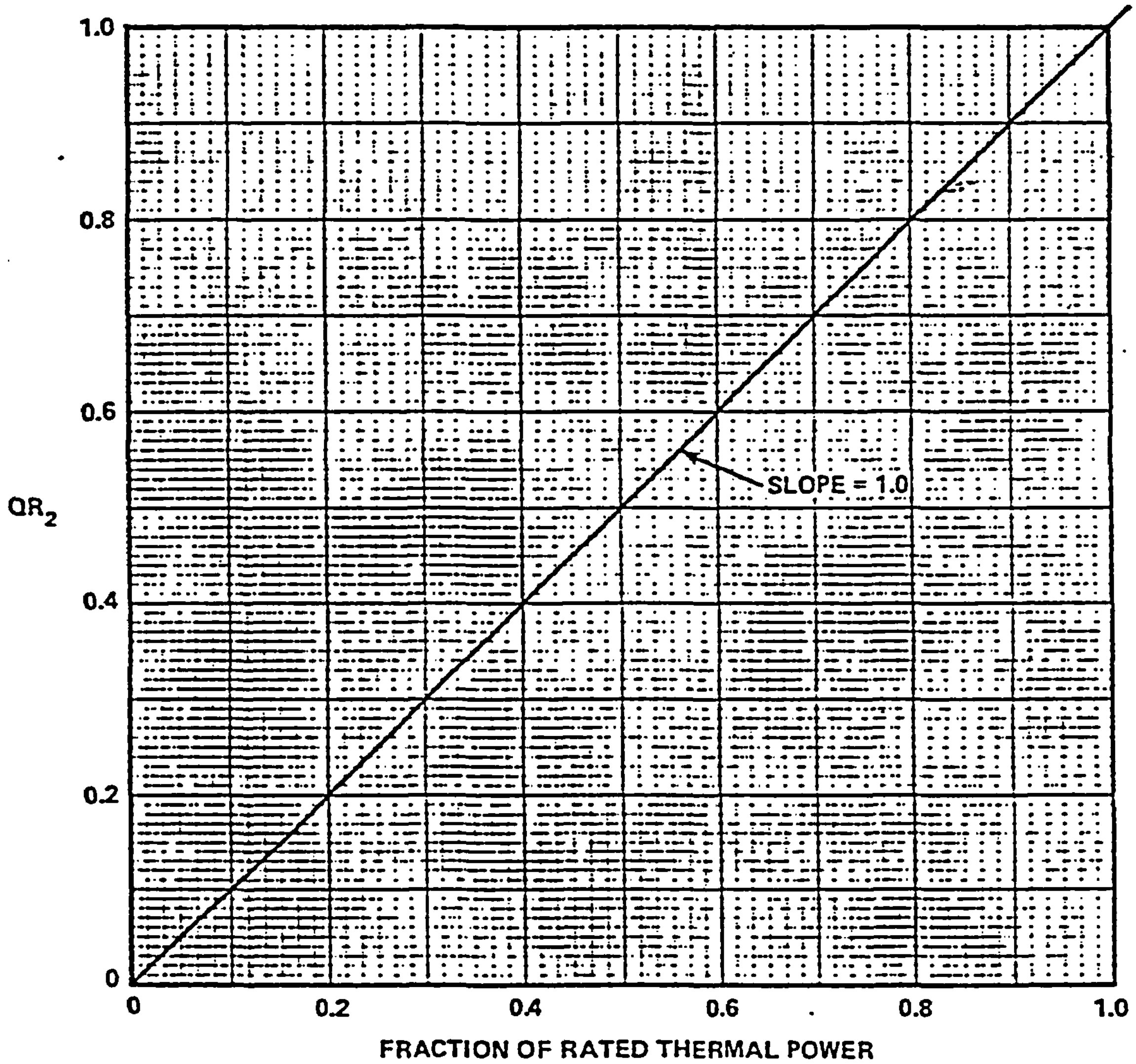


FIGURE 2.2-1

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

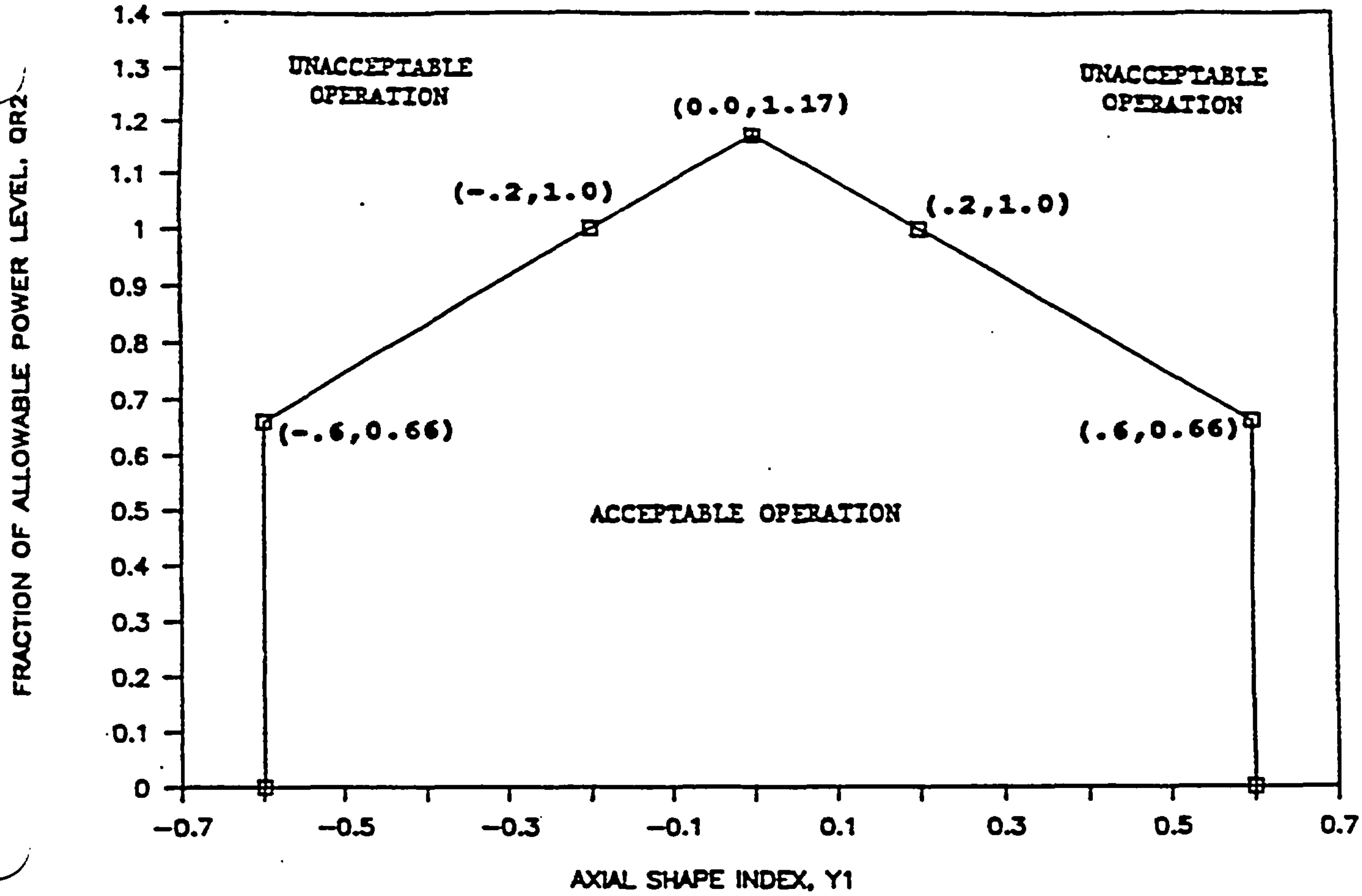


FIGURE 2.2-2

LOCAL POWER DENSITY- HIGH TRIP SETPOINT PART 2 (QR2 Versus Y1)

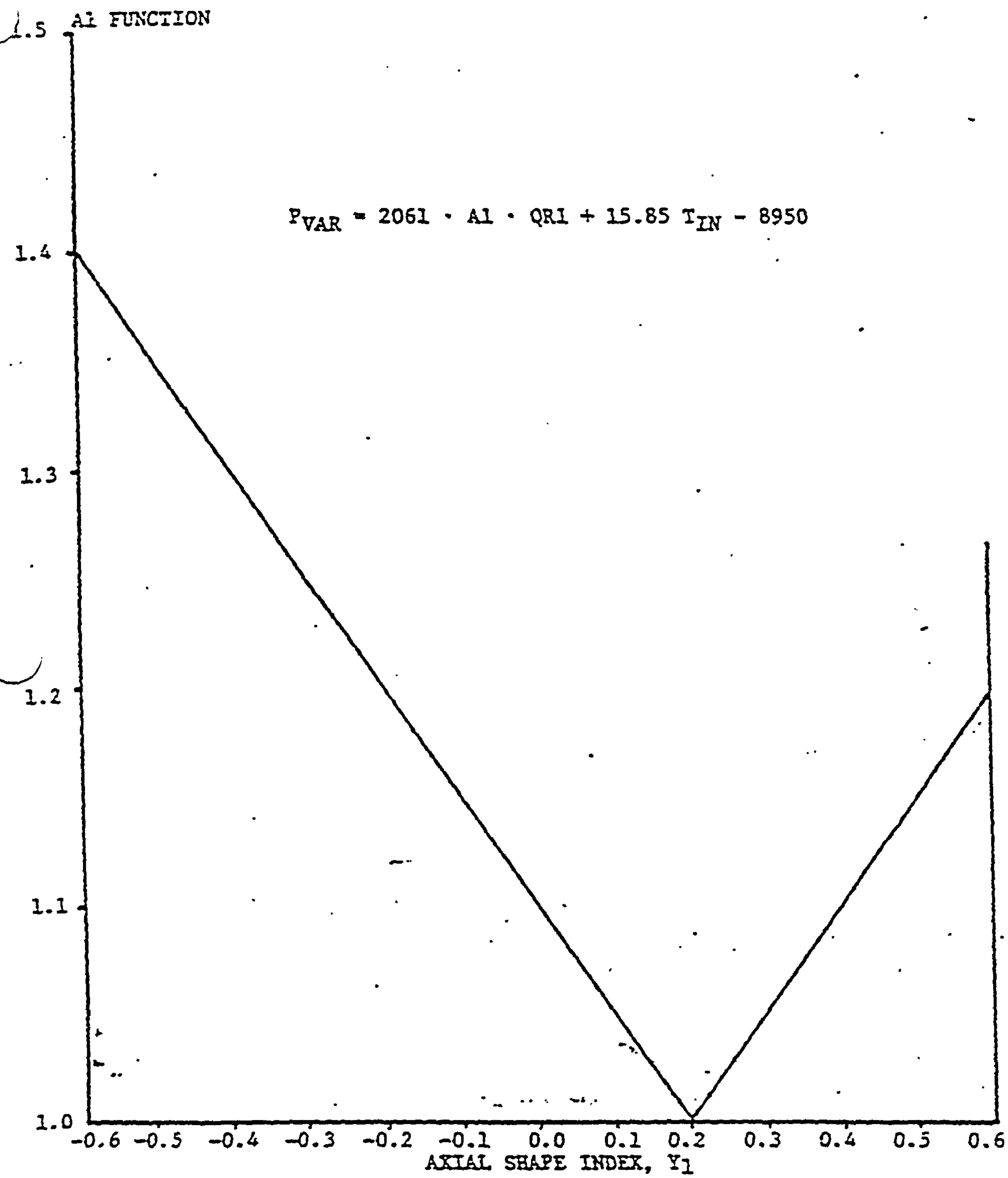


FIGURE 2.2-3

Thermal Margin/Low Pressure Trip Setpoint

$P_{VAR} = 2061 \cdot A1 \cdot QR1 + 15.85\ T_{IN} - 8950$

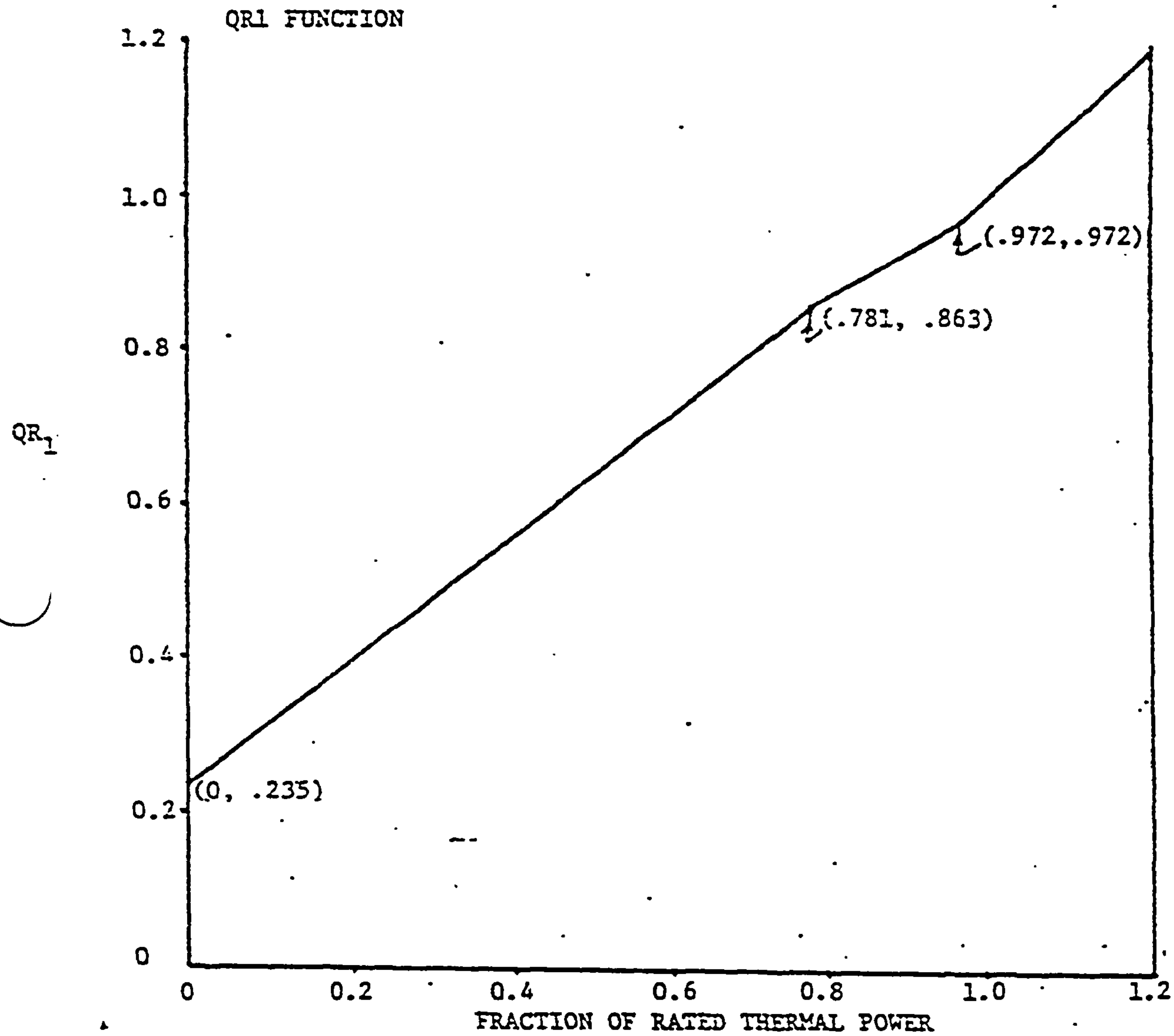


FIGURE 2.2-4

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

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation (LCO) 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(s), completion of the ACTIONS is not required, unless otherwise stated.

3.0.3 When a Limiting Condition for Operation (LCO) 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 specification does not apply by placing it, as applicable in:

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

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

This specification is not applicable in MODES 5 or 6.

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

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

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

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

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

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

APPLICABILITY

LIMITING CONDITION FOR OPERATION (continued)

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 6.8.4.s, "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 ACTIONS for a supported system, the applicable ACTIONS shall be entered in accordance with LCO 3.0.2.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement. Failure to meet a Surveillance Requirement, whether such failure is experienced during the performance of the Surveillance Requirement or between performances of the Surveillance Requirement, shall be failure to meet the LCO. Failure to perform a Surveillance Requirement within the allowed surveillance interval shall be failure to meet the LCO except as provided in SR 4.0.3. Surveillance Requirements do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

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

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 taken.

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 Surveillance Requirement 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

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

4.0.5 Surveillance Requirements for inservice 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 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

b. deleted

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

c. deleted

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

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200 °F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

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

ACTION:

With the SHUTDOWN MARGIN not within limits immediately initiate and continue boration at $\geq$ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the COLR limits:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2## at least once during CEA withdrawal and in accordance with the Surveillance Frequency Control Program until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

---

* See Special Test Exception 3.10.1.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Reactor coolant system boron concentration,
2. CEA 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 $\pm$ 1000 pcm in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

* For Modes 3 and 4, during calculation of shutdown margin with all CEA's <u>verified</u> fully inserted, the single CEA with the highest reactivity worth need <u>not</u> be assumed to be stuck in the fully withdrawn position.

REACTIVITY CONTROL SYSTEMS

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

Within the limits specified in the COLR, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

APPLICABILITY: MODE 5.

ACTION:

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at $\geq$ 40 gpm of greater than or equal to 1900 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

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

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

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

---

* Breaker racked-out.

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be $\geq$ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant to the reactor pressure vessel < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be determined to be $\geq$ 3000 gpm within one hour prior to the start of and in accordance with the Surveillance Frequency Control Program during a reduction in the Reactor Coolant System boron concentration by either:

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

b. Verifying that at least one low pressure safety injection pump is in operation and supplying $\geq$ 3000 gpm to the reactor pressure vessel.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the COLR. The maximum positive limit shall be:

a. Less positive than +7 pcm/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, and

b. Less positive than +2 pcm/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER.

APPLICABILITY: MODES 1 AND 2*#.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 Verify MTC is within the upper limit specified in LCO 3.1.1.4.

a. Prior to entering MODE 1 after each fuel loading, and

b. Each fuel cycle within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup.**

---

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

** Only required to be performed when MTC determined prior to entering MODE 1 is verified using adjusted predicted MTC.

\# See Special Test Exception 3.10.2.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

4.1.1.4.2*** Verify MTC is within the lower limit specified in the COLR.****

Each fuel cycle within 7 EFPD of reaching 2/3 of expected core burnup.

*** If MTC is more negative than the lower limit specified in the COLR when extrapolated to the end of cycle, 4.1.1.4.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit.

**** Only required if the MTC determined in SR 4.1.1.4.1 is not within ±1.6 pcm/°F of the corresponding design value.

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 515°F when the reactor is critical.

APPLICABILITY: MODES 1 and 2#.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. In accordance with the Surveillance Frequency Control Program when the reactor is critical and the Reactor Coolant System temperature (T_{avg}) is < 525°F.

With $K_{eff} \geq 1.0$.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATHS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A flow path from the boric acid makeup tank via either a boric acid pump or a gravity feed connection and any charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump* to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. 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.

---

* The flow path from the RWT to the RCS via a single HPSI pump shall only be established if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case, all charging pumps shall be disabled.

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

SURVEILLANCE REQUIREMENTS

b. At least once per 24 hours, when the Reactor Auxiliary Building air temperature is less than 55°F, by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F, when the flowpath from the Boric Acid Makeup Tank is required to be OPERABLE.

DELETED

THIS PAGE INTENTIONALLY LEFT BLANK

REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

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

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

OR

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

d. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

e. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

f. The flow path from the refueling water storage tank, via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.2 at 200°F; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths 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 during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation Signal.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is below 55°F by verifying that the solution temperature of the Boric Acid Makeup Tank(s) is above 55°F.

CHARGING PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or high pressure safety injection pump* in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump* OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes** until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least one of the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2571 ft. when tested pursuant to the INSERVICE TESTING PROGRAM.

---

* The flow path from the RWT to the RCS via a single HPSI pump shall be established only if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case, all charging pumps shall be disabled.

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

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying that each pump develops a flow rate or greater than or equal to 40 gpm when tested pursuant to the INSERVICE TESTING PROGRAM.

BORIC ACID PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid pump shall be OPERABLE if only the flow path through the boric acid pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one boric acid pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required boric acid pump shall be demonstrated OPERABLE by verifying that the pump develops the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

---

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

REACTIVITY CONTROL SYSTEMS

BORIC ACID PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid pump required for boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid pump 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

4.1.2.6 The above required boric acid pump(s) shall be demonstrated OPERABLE by verifying that the pump(s) develop the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One boric acid makeup tank with a minimum borated water volume of 3650 gallons of 3.0 to 3.5 weight percent boric acid (5245 to 6119 ppm boron).

b. The refueling water tank with:

1. A minimum contained volume of 125,000 gallons,

2. A minimum boron concentration of 1900 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

1. Verifying the boron concentration of the water,

2. Verifying the water level of the tank, and.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the site ambient air temperature is < 40°F.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F when that Boric Acid Makeup Tank is required to be OPERABLE.

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

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following four borated water sources shall be OPERABLE:

a. Boric Acid Makeup Tank 1A in accordance with Figure 3.1-1.

b. Boric Acid Makeup Tank 1B in accordance with Figure 3.1-1.

c. Boric Acid Makeup Tanks 1A and 1B with a minimum combined contained borated water volume in accordance with Figure 3.1-1.

d. The refueling water tank with:

1. A minimum contained volume of 477,360 gallons of water,

2. A minimum boron concentration of 1900 ppm,

3. A maximum solution temperature of 100°F,

4. A minimum solution temperature of 55°F when in MODES 1 and 2, and

5. A minimum solution temperature of 40°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.2 at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

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

1. Verifying the boron concentration of the water source,

REACTIVITY CONTROL SYSTEMS

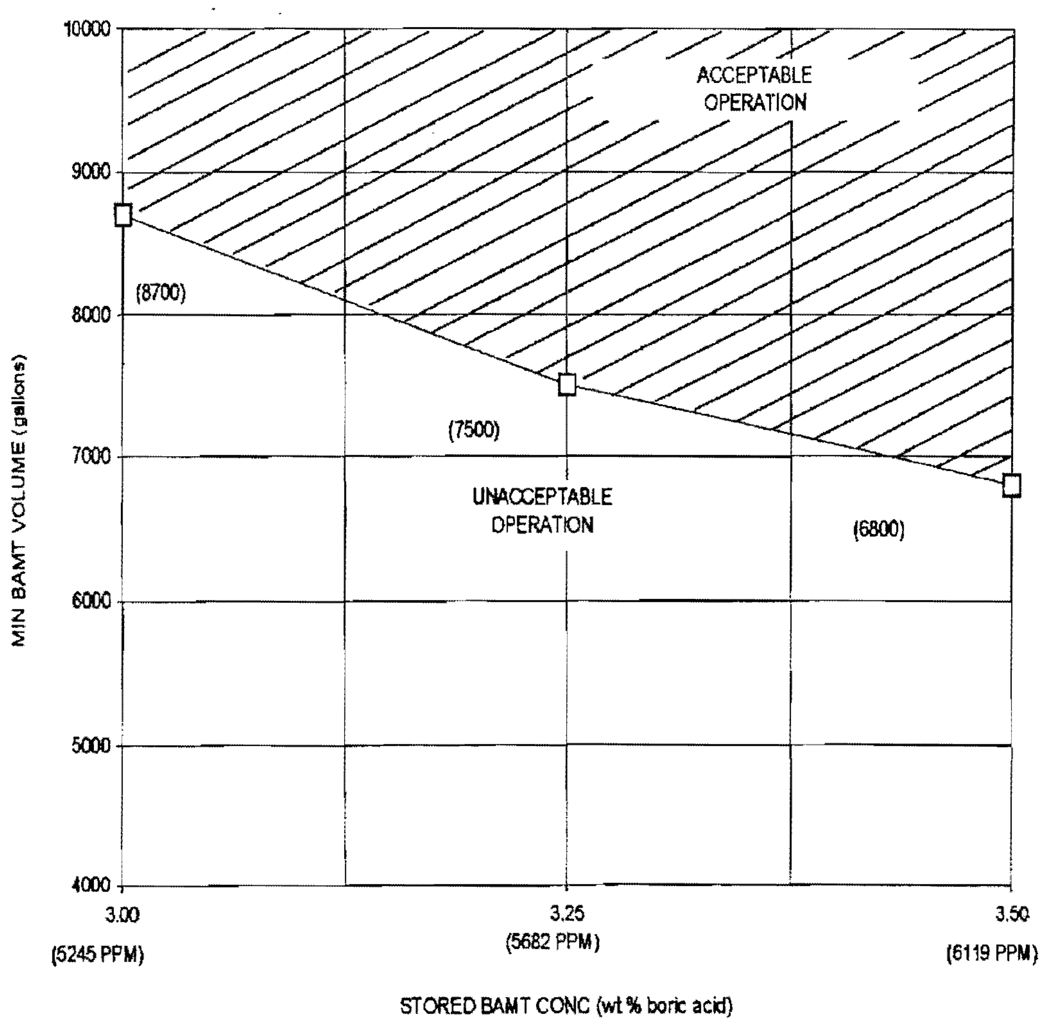
SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the water level in each water source.

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

c. At least once per 24 hours by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F when the Reactor Auxiliary Building air temperature is below 55°F.

**FIGURE 3.1-1 ST. LUCIE 1 MIN BAMT VOLUME
VS STORED BAMT CONCENTRATION
(MODES 1, 2, 3 and 4)**



3/4.1.3 MOVABLE CONTROL ASSEMBLIES

FULL LENGTH CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 The CEA Block Circuit and all full length (shutdown and regulating) CEAs shall be OPERABLE with each CEA of a given group positioned within 7.5 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

b. With the CEA Block Circuit inoperable, within 6 hours either:

   1. With one CEA position indicator per group inoperable, take action per Specification 3.1.3.3, or

   2. With the group overlap and/or sequencing interlocks inoperable, maintain CEAs in groups 3, 4, 5 and 6 fully withdrawn and withdraw the CEAs in group 7 to less than 5% insertion and place and maintain the CEA drive system mode switch in either the "Manual" or "Off" position, or

   3. Be in at least HOT STANDBY.

c. With one full length CEA inoperable due to causes other than addressed by Action a above, but within its above specified alignment requirements and either fully withdrawn or within the long term steady state insertion limits if in CEA group 7, operation in MODES 1 and 2 may continue.

d. With one or more full length CEAs misaligned from any other CEAs in its group by more than 7.5 inches but less than 15 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

*See Special Test Exceptions 3.10.2 and 3.10.5.

LIMITING CONDITION FOR OPERATION (continued)

2. Declared inoperable and satisfy SHUTDOWN MARGIN requirements of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided all of the following conditions are met:

   a) Within 1 hour, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

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

Otherwise, be in at least HOT STANDBY within the next 6 hours.

e. With one full length CEA misaligned from any other CEA in its group by 15 or more inches, operation in MODES 1 and 2 may continue provided that the misaligned CEA is positioned within 7.5 inches of other CEAs in its group in accordance with the time constraints shown in COLR Figure 3.1-1a.

f. With one full-length CEA misaligned from any other CEA in its group by 15 or more inches beyond the time constraints shown in COLR Figure 3.1-1a, reduce power to ≤ 70% of RATED THERMAL POWER prior to completing ACTION f.1 or f.2.

   1. Restore the CEA to OPERABLE status within its specified alignment requirements, or

   2. Declare the CEA inoperable and satisfy the SHUTDOWN MARGIN requirements of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within 1 hour, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

FULL LENGTH CEA POSITION (continued)

LIMITING CONDITION FOR OPERATION (continued)

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

Otherwise, be at least HOT STANDBY within the next 6 hours.

g. With more than one full length CEA inoperable or misaligned from any other CEA in its group by 15 inches (indicated position) or more, be in HOT STANDBY within 6 hours.

h. With one full length CEA inoperable due to causes other than addressed by ACTION a above, and inserted beyond the long term steady state insertion limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length CEA shall be determined to be within 7.5 inches (indicated position) of all other CEAs in its group in accordance with the Surveillance Frequency Control Program except during time intervals when the Deviation Circuit and/or CEA Block Circuit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full length CEA not fully inserted shall be determined to be OPERABLE by inserting it at least 7.5 inches in accordance with the Surveillance Frequency Control Program.

4.1.3.1.3 The CEA Block Circuit shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by a functional test which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).

4.1.3.1.4 The CEA Block Circuit shall be demonstrated OPERABLE by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2:

*a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 92 days, and

b. In accordance with the Surveillance Frequency Control Program.

---

* The licensee shall be excepted from compliance during the startup test program for an entry into MODE 2 from MODE 3 made in association with a measurement of power defect.

DELETED

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All shutdown and regulating CEA reed switch position indicator channels and CEA pulse counting position indicator channels shall be OPERABLE and capable of determining the absolute CEA positions within ± 2.25 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. Deleted.

b. With a maximum of one reed switch position indicator channel per group or one (except as permitted by ACTION item d. below) pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, within 6 hours either:

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

2. Be in HOT STANDBY, or

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

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

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS (Continued)

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 4.5 inches in accordance with the Surveillance Frequency Control Program except during time intervals when the Deviation circuit is inoperable, then compare the pulse counting position indicator and reed switch position indicator channels at least once per 4 hours.

REACTIVITY CONTROL SYSTEMS

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

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

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

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

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

c. In accordance with the Surveillance Frequency Control Program.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, within one hour either:

a. Withdraw the CEA to at least 129.0 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. In accordance with the Surveillance Frequency Control Program.

---

* See Special Test Exception 3.10.2.

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

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits specified in the COLR (regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches) with CEA insertion between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits restricted to:

a. ≤ 4 hours per 24 hour interval,

b. ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. ≤ 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Power Dependent Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

1. Restore the regulating CEA groups to within the limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position and insertion limits specified in the COLR.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits for intervals > 4 hours per 24 hour interval, except during operation pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits are not exceeded, or

2. Any subsequent increase in THERMAL POWER is restricted to ≤ 5% of RATED THERMAL POWER per hour.

* See Special Test Exceptions 3.10.2 and 3.10.5.
\# With $K_{eff} \geq 1.0$.

LIMITING CONDITION FOR OPERATION

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, except during operations pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

2. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Power Dependent Insertion Limits in accordance with the Surveillance Frequency Control Program except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits shall be determined in accordance with the Surveillance Frequency Control Program.

DELETED

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of COLR Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

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

4.2.1.3 Excore Detector Monitoring System – The excore detector monitoring system may be used for monitoring the linear heat rate by:

a. Verifying in accordance with the Surveillance Frequency Control Program that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

b. Verifying in accordance with the Surveillance Frequency Control Program that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on COLR Figure 3.2-2.

SURVEILLANCE REQUIREMENTS (continued)

c. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of COLR Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:

M x N

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F_r^T curve of COLR Figure 3.2-3.

4.2.1.4 Incore Detector Monitoring System# – The incore detector monitoring system may be used for monitoring the linear heat rate by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated in accordance with the Surveillance Frequency Control Program in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on COLR Figure 3.2-1.

If the incore system become inoperable, reduce power to M x N within 4 hours and monitor linear heat rate in accordance with Specification 4.2.1.3.

Pages 3/4 2-4 (Amendment 106), 3/4 2-5 (Amendment 63), and 3/4 2-6 through 3/4 2-8 (Amendment 109) have been deleted from the Technical Specifications. The next page is 3/4 2-9.

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T not within limits, within 6 hours either:

a. Be in at least HOT STANDBY, or

b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of COLR Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6. The THERMAL POWER limit determined from COLR Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on COLR Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by COLR Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of COLR Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r\,(1 + T_q)$ when F_r is calculated with a non-full core power distribution analysis code and shall be calculated as $F_r^T = F_r$ when calculations are performed with a full core power distribution analysis code. F_r^T shall be determined to be within its limit at the following intervals.

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

b. In accordance with the Surveillance Frequency Control Program in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.03.

* See Special Test Exception 3.10.2

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is made using a non-full core power distribution analysis code. The value of T_q used to determine F_r^T in this case shall be the measured value of T_q.

POWER DISTRIBUTION LIMITS

AZIMUTHAL POWER TILT – T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.03.

APPLICABILITY: MODE 1*.

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > .030 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T) is within the limits of Specification 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F_r^T) is within the limits of Specification 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt in accordance with the Surveillance Frequency Control Program when the Subchannel Deviation Alarm is OPERABLE,

---

* See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Calculating the tilt at least once per 12 hours when the Subchannel Deviation Alarm is inoperable, and

c. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Cold Leg Temperature as shown on Table 3.2-1 of the COLR,

b. Pressurizer Pressure* as shown on Table 3.2-1 of the COLR,

c. Reactor Coolant System Total Flow Rate - greater than or equal to 375,000 gpm, and

d. AXIAL SHAPE INDEX as shown on Figure 3.2-4 of the COLR.

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 $\leq$ 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the DNB related parameters shall be verified to be within their limits by instrument readout in accordance with the Surveillance Frequency Control Program.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement** in accordance with the Surveillance Frequency Control Program.

---

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

** Not required to be performed until THERMAL POWER is $\geq$ 90% of RATED THERMAL POWER.

Relocated to the COLR

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

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

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. Power Level – High	4	2(a)	3(f)	1, 2	2
3. Reactor Coolant Flow – Low	4/SG	2(a)/SG	3/SG	1, 2 (e)	2
4. Pressurizer Pressure – High	4	2	3	1, 2	2
5. Containment Pressure – High	4	2	3	1, 2	2
6. Steam Generator Pressure – Low	4/SG	2(b)/SG	3/SG	1, 2	2
7. Steam Generator Water Level – Low	4/SG	2/SG	3/SG	1, 2	2
8. Local Power Density – High	4	2(c)	3	1	2
9. Thermal Margin/Low Pressure	4	2(a)	3	1, 2 (e)	2
9a. Steam Generator Pressure Difference – High	4	2(a)	3	1, 2 (e)	2
10. Loss of Turbine – Hydraulic Fluid Pressure - Low	4	2(c)	3	1	2

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating -- Rate of Change of Power – High	4	2(d)	3	1**, 2 and *	2
b. Shutdown	4	0	2	3, 4, 5	3
12. Reactor Protection System Logic	4	2	4	1, 2*	4
13. Reactor Trip Breakers	4	2	4	1, 2*	4

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

** Mode 1 applicable only when Power Range Neutron Flux power ≤ 15% of RATED THERMAL POWER.

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq$ 1% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}$% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(e) Deleted.

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

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

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

b. Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on than channel provided the other inoperable channel is placed in the tripped condition.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

DELETED

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N/A	N.A.	S/U(1)	N/A
2. Power Level – High				
a. Nuclear Power	SFCP	SFCP(2), SFCP(3), SFCP(5)	SFCP	1,2
b. ΔT Power	SFCP	SFCP(4), SFCP	SFCP	1
3. Reactor Coolant Flow – Low	SFCP	SFCP	SFCP	1, 2
4. Pressurizer Pressure – High	SFCP	SFCP	SFCP	1, 2
5. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2
6. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2
7. Steam Generator Water Level – Low	SFCP	SFCP	SFCP(6, 7)	1, 2
8. Local Power Density – High	SFCP	SFCP	SFCP	1
9. Thermal Margin/Low Pressure	SFCP	SFCP	SFCP	1, 2
9a. Steam Generator Pressure Difference – High	SFCP	SFCP	SFCP	1, 2
10. Loss of Turbine -- Hydraulic Fluid Pressure – Low	N.A.	N.A.	S/U(1)	N.A.
11. Wide Range Logarithmic Neutron Flux Monitor	SFCP	N.A.	S/U(1)	1, 2, 3, 4, 5 and *
12. Reactor Protection System Logic	N.A.	N.A.	SFCP and S/U(1)	1, 2 and *
13. Reactor Trip Breakers	N.A.	N.A.	SFCP	1, 2 and *

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

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

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

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

(4) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

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

(6) - If the as-found setpoint is either outside its predefined as-found acceptance criteria band or is not conservative with respect to the Allowable Value, then the channel shall be declared inoperable and shall be evaluated to verify that it is functioning as required before returning the channel to service.

(7) - The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Field Trip Setpoint, otherwise that channel shall not be returned to OPERABLE status. The Field Trip Setpoint and the methodology used to determine the Field Trip Setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR Section 7.2.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

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

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High	4	2	3	1, 2, 3	9
c. Pressurizer Pressure – Low	4	2	3	1, 2, 3(a)	9
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High-High	4	2(b)	3	1, 2, 3	10A, 10B
3. CONTAINMENT ISOLATION (CIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure – High	4	2	3	1, 2, 3	9
c. Containment Radiation – High	4	2	3	1, 2, 3, 4	9
d. SIAS	---------------------------- (See Functional Unit 1 above) -------------------------------				
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3, 4	8
b. Steam Generator Pressure – Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	9

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	13
6. LOSS OF POWER					
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12
c. 480 V Emergency Bus Under-voltage (Degraded Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12
7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	11
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	11
c. SG Level (1A/1B) - Low	4/SG	2/SG	3/SG	1, 2, 3	14a, 14b, 15
8. AUXILIARY FEEDWATER ISOLATION					
a. SG 1A – SG 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	14a, 14b, 15
b. Feedwater Header 1A – 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	14a, 15

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is $\geq$ 1725 psia.

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

(c) Trip function may be bypassed in this MODE below 685 psig; bypass shall be automatically removed at or above 685 psig.

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 10A - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

ACTION 10B - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 11 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-3 (continued)

TABLE NOTATION

ACTION 13 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If OPERABILITY cannot be restored within 48 hours or in accordance with the Risk Informed Completion Time Program, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

ACTION 14 - With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If an inoperable SG level channel can not be restored to OPERABLE status within 48 hours, then AFAS-1 or AFAS-2 as applicable in the inoperable channel shall be placed in the bypassed condition. If an inoperable SG DP or FW Header DP channel can not be restored to OPERABLE status within 48 hours, then both AFAS-1 and AFAS-2 in the inoperable channel shall be placed in the bypassed condition. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also bypassed or tripped.

ACTION 15 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of the inoperable channels to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 5 psig	≤ 5 psig
c. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1600 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 10 psig	≤ 10 psig
3. CONTAINMENT ISOLATION (CIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 5 psig	≤ 5 psig
c. Containment Radiation - High	≤ 10 R/hr	≤ 10 R/hr
d. SIAS	----- (See FUNCTIONAL UNIT 1 above) ----	
4. MAIN STEAM LINE ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 585 psig	≥ 585 psig
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	48 inches above tank bottom	48 inches above tank bottom

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	≥ 2900 volts with a 1 ± .5 second time delay	≥ 2900 volts with a 1 ± .5 second time delay
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3831 volts with a 18 ± 2 second time delay	≥ 3831 volts with a 18 ± 2 second time delay
c. 480 volts Emergency Bus Undervoltage (Degraded Voltage)	≥ 415 volts with a ≤ 9 second time delay	≥ 415 volts with a ≤ 9 second time delay
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. SG 1A & 1B Level Low	≥ 19.0%	≥ 18.0 %
8. AUXILIARY FEEDWATER ISOLATION		
a. Steam Generator ΔP – High	≤ 275 psid	89.2 to 281 psid
b. Feedwater Header High ΔP	≤ 150.0 psid	56.0 to 157.5 psid

DELETED

DELETED

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3
c. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Containment Pressure –— High-High	SFCP	SFCP	SFCP	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
3. CONTAINMENT ISOLATION (CIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Containment Pressure – High	SFCP	SFCP	SFCP	1, 2, 3
c. Containment Radiation – High	SFCP	SFCP	SFCP	1, 2, 3, 4
d. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
e. SIAS	N.A.	N.A.	SFCP	N.A.
4. MAIN STEAM LINE ISOLATION (MSIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3
5. CONTAINMENT SUMP RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Refueling Water Storage Tank – Low	SFCP	SFCP	SFCP	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP(1)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3
c. 480 V Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1, 2, 3
7. AUXILIARY FEEDWATER (AFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3
b. SG Level (A/B) – Low	SFCP	SFCP	SFCP	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP	1, 2, 3
8. AUXILIARY FEEDWATER ISOLATION				
a. SG Level (A/B) – Low and SG Differential Pressure (BtoA/AtoB) – High	N.A.	SFCP	SFCP	1, 2, 3
b. SG Level (A/B) – Low and Feedwater Header Differential Pressure (BtoA/AtoB) – High	N.A.	SFCP	SFCP	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The logic circuits shall be tested manually in accordance with the Surveillance Frequency Control Program.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

4.3.3.2 In accordance with the Surveillance Frequency Control Program, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area	1	*	≤ 15 mR/hr	$10^{-1} - 10^{4}$ mR/hr	13
b. Containment (CIS)	3	****	≤ 90 mR/hr	$1 - 10^{5}$ mR/hr	16
c. Containment Area – Hi Range	1	1, 2, 3, & 4	≤ 10 R/hr	$1 - 10^{7}$ R/hr	15
d. Control Room Isolation	1 per intake	ALL MODES	≤ 320 cpm	$10 - 10^{7}$ cpm	17
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{6}$ cpm	14
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{6}$ cpm	14

* With fuel in the storage pool or building.

**** During movement of recently irradiated fuel assemblies within containment.

DELETED

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 12 - DELETED

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s),and

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

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION requirements of Specification 3.9.9.

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

DELETED

DELETED

PAGE 3/4 3-26 (ORIGINAL) HAS BEEN DELETED FROM THE TECHNICAL SPECIFICATIONS. THE NEXT PAGE IS 3/4 3-27.

Pages 3/4 3-28 through 3/4 3-32 have been DELETED.

The next page is 3/4 3-33.

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

a. Restore the inoperable channel to OPERABLE status within 30 days, or

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

SURVEILLANCE REQUIREMENTS

NOTE

CHANNEL CALIBRATION is not applicable to the reactor trip breaker indication.

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

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	SWGR	OPEN-CLOSE	1/trip breaker
2. Pressurizer Pressure	Hot Shutdown Panel	1500-2500 psia	1
3. Pressurizer Level	Hot Shutdown Panel	0-100%	1
4. Main Steam Pressure	Hot Shutdown Panel	0-1200 psig	1/steam generator
5. Steam Generator Level	Hot Shutdown Panel	0-100%	1/steam generator
6. Cold Leg Temperature	Hot Shutdown Panel	0-600°F	1

DELETED

THIS PAGE HAS BEEN DELETED

Pages 3/4 3-38 through 3/4 3-40 (Amendment No. 115) have been deleted from the Technical Specifications. The next page is 3/4 3-41.

INSTRUCTIONS

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. Actions per Table 3.3-11.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Pressurizer Water Level	2	1	1, 6
2. Auxiliary Feedwater Flow Rate	1/pump	1/pump	7
3. RCS Subcooling Margin Monitor	2	1	1, 6
4. PORV Position Indicator Acoustic Flow Monitor	1/valve	1/valve	2
5. PORV Block Valve Position Indicator	1/valve	1/valve	2
6. Safety Valve Position Indicator	1/valve	1/valve	3
7. Incore thermocouples	4/core quadrant	2/core quadrant	1, 6
8. Containment Sump Water Level (Narrow Range)	1*	1*	4, 5
9. Containment Sump Water Level (Wide Range)	2	1	4, 5
10. Reactor Vessel Level Monitoring System	2**	1**	4, 5
11. Containment Pressure	2	1	1, 6

* The non-safety grade containment sump water level instrument may be substituted.

** Definition of OPERABLE: A channel is composed of eight (8) sensors in a probe, of which four (4) sensors must be OPERABLE.

TABLE 3.3-11 (continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the Total No. of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 2 - With position indication inoperable, restore the inoperable indicator to OPERABLE status or close the associated PORV block valve and remove power from its operator within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information once per shift to determine valve position.

ACTION 4 - With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to the specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 5 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

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

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

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

ACTION 6 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 7 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

DELETED

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 The reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant loops shall be in operation.*

a. Reactor Coolant Loop A and its associated steam generator and at least one associated reactor coolant pump.

b. Reactor Coolant Loop B and its associated steam generator and at least one associated reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

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

b. With no reactor coolant loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.1 and within one (1) hour initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

4.4.1.2.3 The required steam generators shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% of narrow range indication in accordance with the Surveillance Frequency Control Program.

* All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.1.1.1 and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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 reactor coolant or shutdown cooling loop shall be in operation.*

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

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

c. Shutdown Cooling Loop A,

d. Shutdown Cooling Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant or shutdown cooling loops OPERABLE, within one (1) hour initiate corrective action to return the required loops to OPERABLE status and immediately initiate action to make at least one steam generator available for decay heat removal via natural circulation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

b. With no reactor coolant or shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.1 and within one (1) hour initiate corrective action to return the required reactor coolant loop to operation.

---

* All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.1.1.1 and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥ 10% of narrow range indication in accordance with the Surveillance Frequency Control Program.

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

4.4.1.3.4 Verify required shutdown cooling train 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

COLD SHUTDOWN – LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation* and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 10% of narrow range indication.

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

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, within one (1) hour initiate corrective action to return the required loops to OPERABLE status or to restore the required level.

b. With no shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and within one (1) hour initiate corrective action to return the required shutdown 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 shutdown cooling 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 shutdown cooling train locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

---

* The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

\# One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

\## A reactor coolant pump shall not be started with two idle loops unless the secondary water temperature of each steam generator is less than 30°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN – LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation*.

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, within one (1) hour initiate corrective action to return the required loops to OPERABLE status.

b. With no shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and within one (1) hour initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling 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 shutdown cooling train locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

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

* The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

DELETED

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of $\geq$ 2422.8 psig and $\leq$ 2560.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 281°F.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the next 6 hours.

b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures $\leq$ 281°F within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 Verify each pressurizer code safety valves is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, as-left lift settings shall be within +/- 1% of 2500 psia.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble, and with at least 150 kw of pressurizer heaters capable of being supplied by emergency power.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.4 In accordance with 4.8.1.1.2.

REACTOR COOLANT SYSTEM

STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the SG Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: *

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

* Separate Action entry is allowed for each SG tube.

THIS PAGE DELETED

THIS PAGE DELETED

THIS PAGE DELETED

SURVEILLANCE REQUIREMENTS (Continued)

THIS PAGE DELETED

THIS PAGE DELETED

REACTOR COOLANT SYSTEM

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 will be OPERABLE:

a. The reactor cavity sump inlet flow monitoring system; and

b. One containment atmosphere radioactivity monitor (gaseous or particulate).

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

ACTION:

a. With the reactor cavity sump inlet flow monitoring system inoperable with an operable containment particulate radioactivity monitor, perform a RCS water inventory balance at least once per 24* hours and restore the sump inlet flow monitoring system to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the reactor cavity sump inlet flow monitoring system inoperable with only the containment gaseous radioactivity monitor operable, perform an RCS water inventory balance at least once per 24* hours and analyze grab samples of the containment atmosphere at least once per 12 hours, and either restore the sump inlet flow monitoring system to OPERABLE status within 7 days or restore the containment particulate radioactivity monitor to OPERABLE status within 7 days and enter action a. above with the time in this action applied against the allowed outage time of action a.; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the required radioactivity monitor inoperable, analyze grab samples of the containment atmosphere or perform a RCS water inventory balance at least once per 24* hours, and restore the required radioactivity monitor to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With all required monitors inoperable, enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The RCS leakage detection instruments shall be demonstrated OPERABLE by:

a. Performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor in accordance with Surveillance Requirement 4.3.3.1.

b. Performance of the CHANNEL CALIBRATION of the reactor cavity sump inlet flow monitoring system in accordance with the Surveillance Frequency Control Program.

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

DELETED

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM 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, and

e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

c. *Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program except when operating in the shutdown cooling mode,

d. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program, and

e. Verifying each Reactor Coolant System Pressure Isolation Valve leakage (Table 3.4.6-1) to be within limits:

   1. Prior to entering MODE 2 after refueling,

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

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

   4. The provision of Specification 4.0.4 is not applicable for entry into MODE 3 or 4.

f. Whenever integrity of a pressure isolation valve listed in Table 3.4.6-1 cannot be demonstrated the integrity of the remaining check valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in each high pressure line having a leaking valve shall be recorded daily; and

g. Primary-to-secondary leakage shall be verified ≤150 gallons per day through any one steam generator in accordance with the Surveillance Frequency Control Program.**

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

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

TABLE 3.4.6-1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Check Valve No.

V3227
V3123
V3217
V3113
V3237
V3133
V3247
V3143
V3124
V3114
V3134
V3144

NOTES

(a) Maximum Allowable Leakage (each valve):

1. Leakage rates less than or equal to 1.0 gpm are acceptable.

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

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

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Mimimum test differential pressure shall not be less than 150 psid.

DELETED

DELETED

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 518.9 μCi/gram DOSE EQUIVALENT XE-133.

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

ACTION:

a. With the specific activity of the primary coolant >1.0 μCi/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is $\leq$ 60.0 μCi/gram once per four hours.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131, but $\leq$ 60.0 μCi/gram DOSE EQUIVALENT I-131, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 μCi/gram limit. LCO 3.0.4.c is applicable.

c. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for greater than 48 hours during one continuous time interval, or > 60.0 μCi/gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With the specific activity of the primary coolant > 518.9 μCi/gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 518.9 μCi/gram DOSE EQUIVALENT XE-133 limit. LCO 3.0.4.c is applicable.

e. With the specific activity of the primary coolant > 518.9 μCi/gram DOSE EQUIVALENT XE-133 for greater than 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performing sampling and analysis as described in Table 4.4-4.

DELETED

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. DOSE EQUIVALENT XE-133 Determination	SFCP	1, 2, 3 and 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	SFCP	1
3. Isotopic Analysis for Iodine Including I-131, I-132, I-133, I-134, and I-135	a) Once per 4 hours, whenever the DOSE EQUIVALENT I-131 exceeds 1.0 μCi/gram, and	1#, 2#, 3#, and 4#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

\# Until the specific activity of the primary coolant system is restored within its limits.

DELETED

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.*

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} to less than 200°F within the following 30 hours in accordance with Figure 3.4-2b.

---

* During hydrostatic testing operations above system design pressure, a maximum temperature change in any one hour period shall be limited to 5°F.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.1

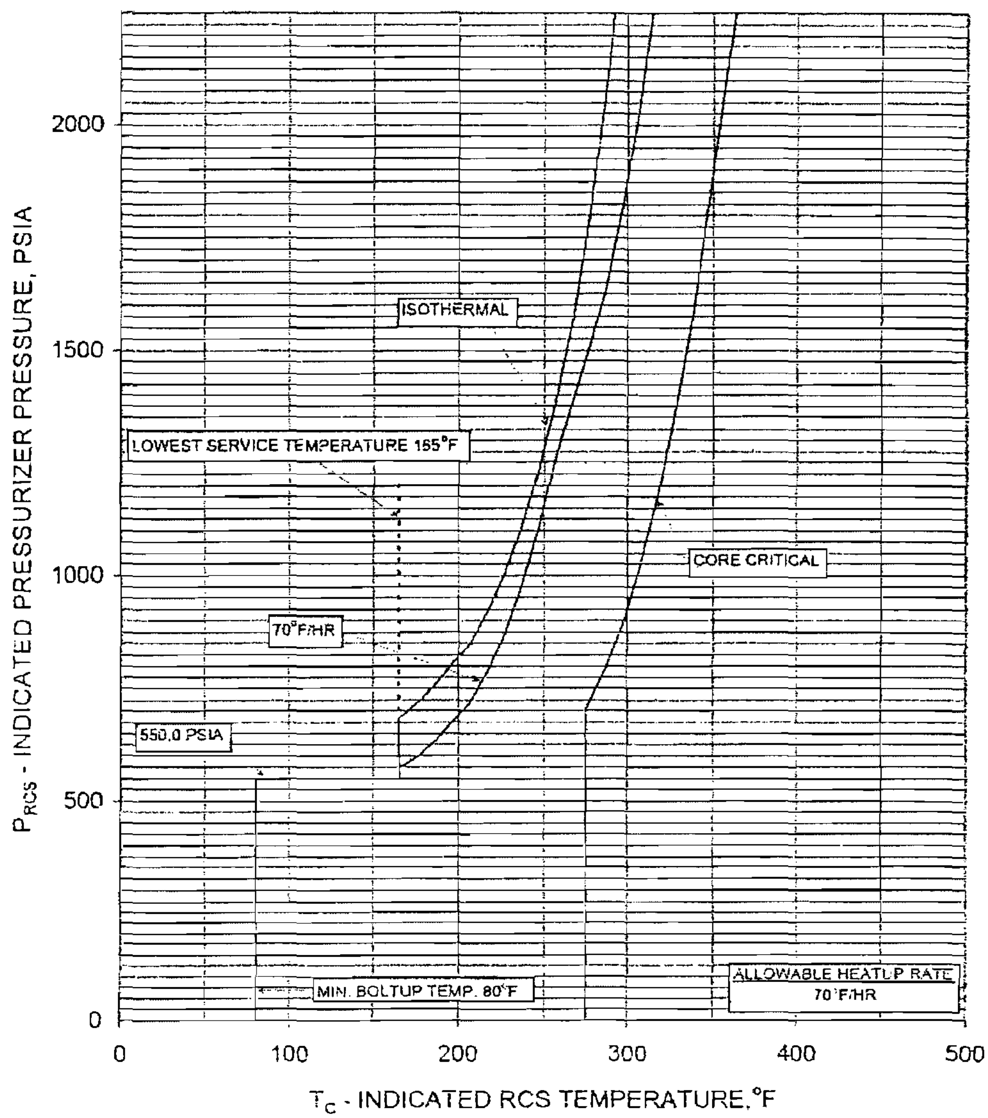
a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update Figures 3.4-2a and 3.4-2b.

FIGURE 3.4-2a

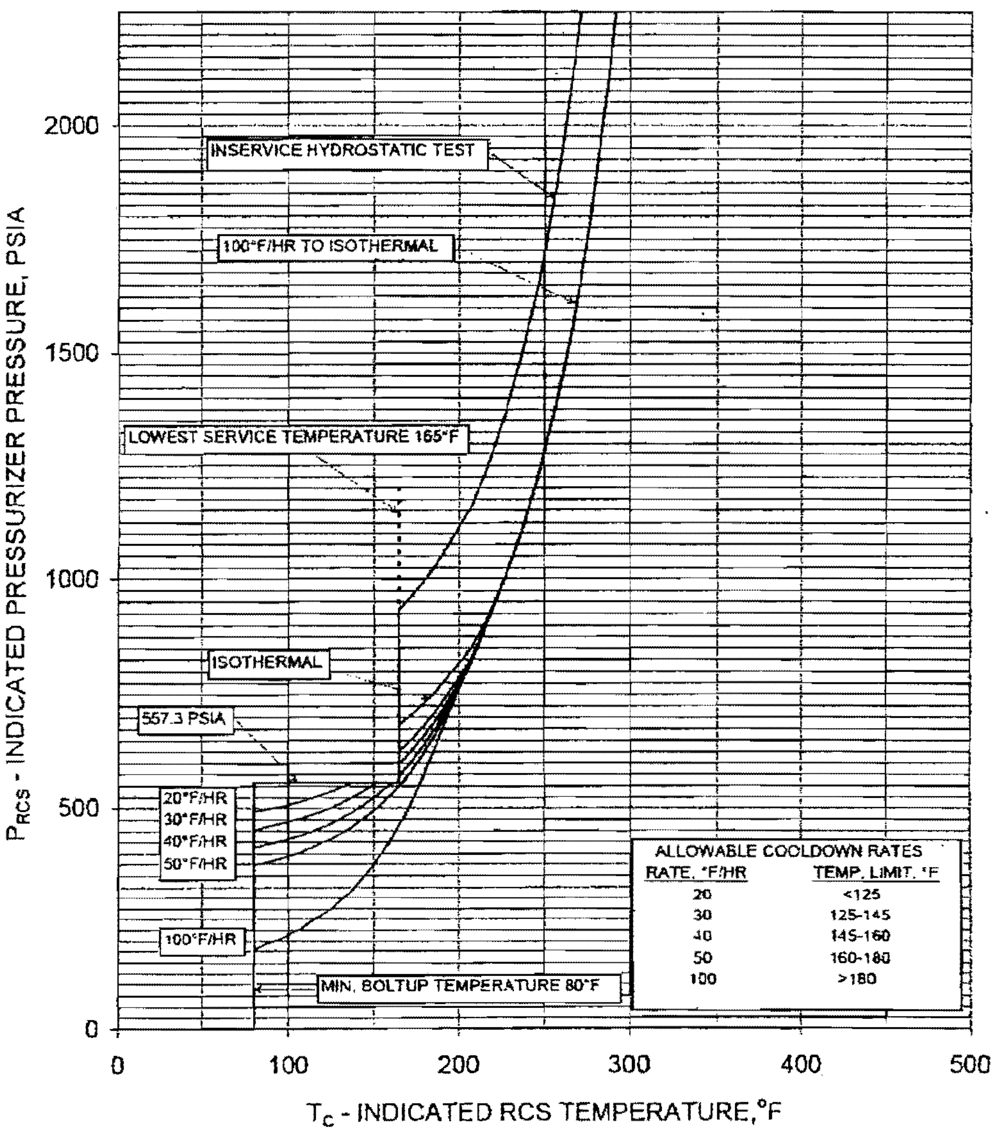
ST. LUCIE UNIT 1 P/T LIMITS, 54 EFPY
HEATUP AND CORE CRITICAL



Limiting Material: Lower Shell Axial Welds (Ht. #305424)
Limiting ART Values at 54 EFPY: 1/4T, 210°F
3/4T, 156°F

FIGURE 3.4-2b

ST. LUCIE UNIT 1 P/T LIMITS, 54 EFPY
COOLDOWN AND INSERVICE TEST



Limiting Material: Lower Shell Axial Welds (Ht. #305424)

Limiting ART Values at 54 EFPY: 1/4T, 210°F
3/4T, 156°F

DELETED

DELETED

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

c. A maximum Reactor Coolant System spray water temperature differential of 350°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 analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psia 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 steady state operation.

DELETED

SURVEILLANCE REQUIREMENTS (Continued)

Pages 3/4 4-28 through 3/4 4-55 (Amendment No. 90), and Pages 3/4 4-56 through 3/4 4-57 (Amendment No. 80) have been deleted from the Technical Specifications. The next page is 3/4 4-58.

REACTOR COOLANT SYSTEM

PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.12 Each Power Operated Relief Valve (PORV) Block Valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one or more block valve(s) inoperable, within 1 hour or in accordance with the Risk Informed Completion Time Program either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.12 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.

REACTOR COOLANT SYSTEM

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.13 Two power operated relief valves (PORVs) shall be OPERABLE, with their setpoints selected to the low temperature mode of operation as follows:

a. A setpoint of less than or equal to 350 psia shall be selected during heatup, cooldown and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 200°F.

b. A setpoint of less than or equal to 530 psia shall be selected during heatup, cooldown and isothermal conditions when the temperature of any RCS cold leg is greater than 200°F and less than or equal to 300°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 300°F, MODE 5, and MODE 6 when the head is on the reactor vessel; and the RCS is not vented through greater than a 1.75 square inch vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days; or depressurize and vent the RCS through greater than a 1.75 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, restore at least one PORV to operable status or complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within 24 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

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

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

SURVEILLANCE REQUIREMENTS

4.4.13 Each PORV shall be demonstrated OPERABLE by:

a. Verifying the PORV isolation valve is open in accordance with the Surveillance Frequency Control Program; and

b. Performance of a CHANNEL FUNCTION TEST, but excluding valve operation, in accordance with the Surveillance Frequency Control Program; and

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMP - STARTING

LIMITING CONDITION FOR OPERATION

3.4.14 If the steam generator temperature exceeds the primary temperature by more than 30°F, the first idle reactor coolant pump shall not be started.

APPLICABILITY: MODES 4# and 5.

ACTION:

If a reactor coolant pump is started when the steam generator temperature exceeds primary temperature by more than 30°F, evaluate the subsequent transient to determine compliance with Specification 3.4.9.1.

SURVEILLANCE REQUIREMENTS

4.4.14 Prior to starting a reactor coolant pump, verify that the steam generator temperature does not exceed primary temperature by more than 30°F.

Reactor Coolant System Cold Leg Temperature is less than 300°F.

REACTOR COOLANT SYSTEM

3/4.4.15 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

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

a. Pressurizer steam space, and

b. Reactor vessel head.

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 paths 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

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

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

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

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS (SITs)

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. Between 1090 and 1170 cubic feet of borated water,

c. A minimum boron concentration of 1900 ppm, and

d. A nitrogen cover-pressure of between 230 and 280 psig.

APPLICABILITY: MODES 1, 2 and 3 with pressurizer pressure ≥ 1750 psia

ACTION:

a. With one SIT inoperable due to boron concentration not within limits, or due to an inability to verify the required water volume or nitrogen cover-pressure, restore the inoperable SIT to OPERABLE status with 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one SIT inoperable due to reasons other than those stated in ACTION-a, restore the inoperable SIT to OPERABLE status within 24 hours ; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

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

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

2. Verifying that each safety injection tank isolation valve is open.

SURVEILLANCE REQUIREMENTS (continued)

b. In accordance with the Surveillance Frequency Control Program and once within 6 hours after each solution volume increase of $\geq$ 1% of tank volume by verifying the boron concentration of the safety injection tank solution. This latter surveillance is not required when the volume increase makeup source is the RWT and the RWT has not been diluted since verifying that the RWT boron concentration is equal to or greater than the safety injection tank boron concentration limit.

c. In accordance with the Surveillance Frequency Control Program when the RCS pressure is above 1750 psia, by verifying that power to the isolation valve operator is removed by maintaining the breaker open under administrative control.

d. In accordance with the Surveillance Frequency Control Program by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When the RCS pressure exceeds 350 psia, and

2. Upon receipt of a safety injection test signal.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) pump,

b. One OPERABLE low-pressure safety injection pump,

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

> **NOTE**
>
> One ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.a or 3.1.2.2.d. The second ECCS subsystem charging pump shall satisfy the flow path requirements of Specification 3.1.2.2.b or 3.1.2.2.e.

d. One OPERABLE charging pump.

APPLICABILITY: MODES 1, 2 and 3 with pressurizer pressure ≥ 1750 psia.

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 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
1. V-3659	1. Mini-flow isolation	1. Open
2. V-3660	2. Mini-flow isolation	2. Open

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

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

2. Verifying ECCS train locations susceptible to gas accumulation are sufficiently filled with water.

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

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

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

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

1. Verifying proper operation of the open permissive interlock (OPI) and the valve open/high SDCS pressure alarms for isolation valves V3651, V3652, V3480, V3481.

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 corrosion.

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

SURVEILLANCE REQUIREMENTS (continued)

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

1. Verifying that each automatic valve in the flow paths actuates to its correct position on a Safety Injection Actuation Signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Signal;

   a. High-Pressure Safety Injection Pumps.

   b. Low-Pressure Safety Injection Pumps.

   c. Charging Pumps.

3. Verifying that upon receipt of an actual or simulated Recirculation Actuation Signal: each low-pressure safety injection pump stops, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.

f. By verifying that each of the following pumps develops the specified total developed head when tested pursuant to the INSERVICE TESTING PROGRAM.

1. High-Pressure Safety Injection pumps.

2. Low-Pressure Safety Injection pumps.

SURVEILLANCE REQUIREMENTS (Continued)

THIS PAGE INTENTIONALLY LEFT BLANK

ECCS SUBSYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. In MODES 3* and 4#, one ECCS subsystem composed of one OPERABLE high pressure safety injection pump and one OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal.

b. Prior to decreasing the reactor coolant system temperature below 270°F a maximum of only one high pressure safety injection pump shall be OPERABLE with its associated header stop valve open.

c. Prior to decreasing the reactor coolant system temperature below 236°F all high pressure safety injection pumps shall be disabled and their associated header stop valves closed except as allowed by Specifications 3.1.2.1 and 3.1.2.3.

APPLICABILITY: MODES 3* and 4.
MODES 5 and 6 when the Pressurizer manway cover is in place and the reactor vessel head is on.

ACTION:

a. With no ECCS subsystems OPERABLE in MODES 3* and 4#, immediately restore one ECCS subsystem to OPERABLE status or be in COLD SHUTDOWN within 20 hours.

b. With RCS temperature below 270°F and with more than the allowed high pressure safety injection pump OPERABLE or injection valves and header isolation valves open, immediately disable the high pressure safety injection pump(s) or close the header isolation valves.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

d. LCO 3.0.4.b is not applicable to ECCS High Pressure Safety Injection subsystem when entering MODE 4.

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 The high pressure safety injection pumps shall be verified inoperable and the associated header stop valves closed prior to decreasing below the above specified Reactor Coolant System temperature and once per month when the Reactor Coolant System is at refueling temperatures.

---

* With pressurizer pressure < 1750 psia.

\# REACTOR COOLANT SYSTEM cold leg temperature above 250°F.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained volume 477,360 gallons of borated water,

b. A minimum boron concentration of 1900 ppm,

c. A maximum water temperature of 100°F,

d. A minimum water temperature of 55°F when in MODES 1 and 2, and

e. A minimum water temperature of 40°F when in MODES 3 and 4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water tank 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 RWT shall be demonstrated OPERABLE:

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

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

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

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT VESSEL

CONTAINMENT VESSEL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be demonstrated:

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

   1. All containment vessel 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 on an intermittent basis under administrative control, and

   2. All containment vessel equipment hatches are closed and sealed.

b. By verifying that each containment vessel air lock is OPERABLE per 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 LEAKAGE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment leakage rate exceeding the acceptance criteria of the Containment Leakage Rate Testing Program, within 1 hour initiate action to be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the overall leakage rate to less than that specified by the Containment Leakage Rate Testing Program prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the required test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

SURVEILLANCE REQUIREMENTS (continued)

Pages 3/4 6-4 through 3/4 6-9 have been DELETED.

Page 3/4 6-10 is the next valid page.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

NOTE
• If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door. No more than one air lock door shall be open at any time.
• Enter the ACTION of LCO 3.6.1.2, "Containment Leakage," when air lock leakage results in exceeding 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 closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

b. With one or both containment air lock(s) inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed in the affected air lock(s) and restore the inoperable air lock(s) to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

SURVEILLANCE REQUIREMENTS (continued)

a. By verifying leakage rates and air lock door seals in accordance with the Containment Leakage Rate Testing Program; and

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

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.490 and +0.5 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

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

Location

a. Containment fan cooler No. 1A air intake, elevation 45 feet.

b. Containment fan cooler No. 1B air intake, elevation 45 feet.

c. Containment fan cooler No. 1C air intake, elevation 62 feet.

d. Containment fan cooler No. 1D air intake, elevation 45 feet.

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, in accordance with the containment Leakage Rate Testing Program, by a visual inspection of the accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: Containment Spray System: MODES 1, 2, and MODE 3 with Pressurizer Pressure ≥ 1750 psia.

Containment Cooling System: MODES 1, 2, and 3.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure ≥ 1750 psia:

   a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 54 hours.

   b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   c. With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.

NOTE

Action not applicable when second containment spray train intentionally made inoperable.

   d. With two containment spray trains inoperable, within 1 hour verify TS 3.7.7, "Control Room Emergency Ventilation System," is met, and restore at least one containment spray train to OPERABLE status within 24 hours; otherwise, be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

   e. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

2. Mode 3 with Pressurizer Pressure < 1750 psia:

   a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within the next 6 hours.

   b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is positioned to take suction from the RWT on a Containment Pressure -- High High test signal.*

b. By verifying that each spray pump develops the specified discharge pressure when tested pursuant to the INSERVICE TESTING PROGRAM.

c. In accordance with the Surveillance Frequency Control Program by verifying containment spray system locations susceptible to gas accumulation are sufficiently filled with water.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

3. Verifying that upon a recirculation actuation signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

e. By verifying each spray nozzle is unobstructed following maintenance which could result in nozzle blockage.

4.6.2.1.1. Each containment cooling train shall be demonstrated OPERABLE:

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

1. Starting each cooling train fan unit from the control room and verifying that each unit operates for at least 15 minutes, and

2. Verifying a cooling water flow rate of greater than or equal to 1200 gpm to each cooling unit.

b. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that each containment cooling train starts automatically on an SIAS test signal.

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 4010 and 5000 gallons of between 28.5 and 30.5% by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2 and 3.*

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, during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a CSAS test signal.

---

* Applicable when pressurizer pressure is $\geq$ 1750 psia.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

d. In accordance with the Surveillance Frequency Control Program by verifying a minimum sodium hydroxide (NaOH) flow rate of 10.5 gpm from the spray additive tank to a drain connection immediately downstream of the tank outlet valve, and a demineralized water flow rate of 18 $\pm$ 1.5 gpm from that same drain connection to each containment spray pump.

DELETED

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves shall be OPERABLE:

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

> **NOTE**
>
> 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 leakage results in exceeding overall containment leakage rate acceptance criteria.

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

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours or in accordance with the Risk Informed Completion Time Program, or

b. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours or in accordance with the Risk Informed Completion Time Program by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 The isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the cycling test, and verification of isolation time.

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Containment Isolation test signal, and/or SIAS test signal, each isolation valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

Pages 3/4 6-21 through 3/4 6-22 have been DELETED.

Page 3/4 6-23 is the next valid page.

DELETED

DELETED

DELETED

CONTAINMENT SYSTEMS

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two vacuum relief lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one vacuum relief line inoperable, restore the vacuum relief line to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Verify each vacuum relief line OPERABLE in accordance with the INSERVICE TESTING PROGRAM.

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent shield building ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one shield building ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

> **NOTE**
> Action not applicable when second shield building ventilation system intentionally made inoperable.

b. With two shield building ventilation systems inoperable, within 1 hour verify at least one train of containment spray is OPERABLE, and restore at least one shield building ventilation system to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each shield building ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 continuous minutes with the heaters on.

b. By performing required shield building ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

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

1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.
2. Verifying that the filtration system starts automatically on a Containment Isolation Signal (CIS).
3. Verifying that the filter cooling makeup air and cross connection valves can be manually opened.
4. Verifying that each system produces a negative pressure of $\geq$ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS).

Page Deleted

Page Deleted

CONTAINMENT SYSTEMS

SHIELD BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.2 SHIELD BUILDING INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without SHIELD BUILDING INTEGRITY, restore SHIELD BUILDING INTEGRITY within 24 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.6.2 SHIELD BUILDING INTEGRITY shall be demonstrated 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.

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the shield building 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

4.6.6.3 The structural integrity of the shield building shall be determined, in accordance with the Containment Leakage Rate Testing Program, by a visual inspection of the accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation.

3/4.7 PLANT SYSTEMS

3.4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 4.7-1.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With both 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 Level-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 Verify each main steam line code safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, as-left lift settings shall be within +/- 1% of 1000 psia for valves 8201 through 8208, and within +/- 1% of 1040 psia for valves 8209 through 8216 specified in Table 4.7-1.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	88.5
2	79.8
3	66.5

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER		LIFT SETTING*
	Header A	Header B	
a.	8201	8205	≥ 955.3 psig and ≤ 1015.3 psig
b.	8202	8206	≥ 955.3 psig and ≤ 1015.3 psig
c.	8203	8207	≥ 955.3 psig and ≤ 1015.3 psig
d.	8204	8208	≥ 955.3 psig and ≤ 1015.3 psig
e.	8209	8213	≥ 994.1 psig and ≤ 1046.1 psig
f.	8210	8214	≥ 994.1 psig and ≤ 1046.1 psig
g.	8211	8215	≥ 994.1 psig and ≤ 1046.1 psig
h.	8212	8216	≥ 994.1 psig and ≤ 1046.1 psig

* +/-3% for valves a through d and +2%/-3% for valves e through h

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor driven feedwater pumps, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump steam supply inoperable, restore the inoperable auxiliary feedwater pump steam supply to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one auxiliary feedwater pump inoperable, restore the auxiliary feedwater pump to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one auxiliary feedwater pump steam supply inoperable and one motor-driven auxiliary feedwater pump inoperable, either restore the inoperable auxiliary feedwater pump steam supply OR restore the inoperable motor-driven auxiliary feedwater pump to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE
LCO 3.0.3 and all other LCO Actions requiring MODE changes are suspended until one AFW pump is restored to OPERABLE status.

e. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status.

f. LCO 3.0.4.b is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

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

SURVEILLANCE REQUIREMENTS (Continued)

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.

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

1. Verifying that each automatic valve in the flowpath actuates to its correct position upon receipt of the Auto Start actuation test signal.

2. Verifying that each auxiliary feedwater pump starts automatically as designed upon receipt of the Auto Start actuation test signal.

c. By verifying the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head when tested in accordance with the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 when testing the steam turbine-driven AFW pump and this Surveillance must be performed within 24 hours after entering MODE 3 and prior to entering MODE 2.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 153,400 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, restore the condensate storage tank to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performing sampling and analysis as described in Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY
1. Gross Activity Determination	SFCP
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

MODES 2 and 3 - With one or both main steam isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may continue provided:

1. The inoperable main steam isolation valves are closed within 8 hours, and
2. The inoperable main steam isolation valves are verified closed once per 7 days.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve that is open shall be demonstrated OPERABLE by verifying full closure within 6.0 seconds when tested pursuant to the INSERVICE TESTING PROGRAM.

Pages 3/4 7-11 through 3/4 7-12 (Amendment No. 86) have been deleted from the Technical Specifications. The next page is 3/4 7-13.

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: ALL MODES.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

b. Perform an analysis to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generators shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant in the steam generators is < 70°F.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

NOTE
Enter applicable ACTIONS of LCO 3.4.1.3, "Reactor Coolant System - Hot Shutdown," for shutdown cooling loops made inoperable by CCW.

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two 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.

b. In accordance with the Surveillance Frequency Control Program during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation Signal.

SURVEILLANCE REQUIREMENTS (Continued)

THIS PAGE INTENTIONALLY LEFT BLANK

PLANT SYSTEMS

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent intake cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

NOTE
Enter applicable ACTIONS of LCO 3.4.1.3, "Reactor Coolant System - Hot Shutdown," for shutdown cooling loops made inoperable by ICW.

With only one intake cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two intake 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.

b. In accordance with the Surveillance Frequency Control Program during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation signal.

SURVEILLANCE REQUIREMENTS (Continued)

THIS PAGE INTENTIONALLY LEFT BLANK

PLANT SYSTEMS

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5.1 The ultimate heat sink shall be OPERABLE with:

a. Cooling water from the Atlantic Ocean providing a water level above −10.5 feet elevation, Mean Low Water, at the plant intake structure, and

b. Two OPERABLE valves in the barrier dam between Big Mud Creek and the intake structure.

APPLICABILITY: At all times.

ACTION:

a. With the water level requirement of the above Specification not satisfied, be in at least HOT STANDBY within six hours and provide cooling water from Big Mud Creek within the next 12 hours.

b. With one isolation valve in the barrier dam between Big Mud Creek and the intake structure inoperable, restore the inoperable valve to OPERABLE status within 72 hours or, within the next 24 hours, install a temporary flow barrier and open the barrier dam isolation valve. The availability of the onsite equipment capable of removing the barrier shall be verified at least once per seven days thereafter.

c. With both of the isolation valves in the barrier dam between the intake structure and Big Mud Creek inoperable, within 24 hours either:

1) Install both temporary flow barriers and manually open both barrier dam isolation valves. The availability of the onsite equipment capable of removing the barriers shall be verified at least once per seven days thereafter, or

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

SURVEILLANCE REQUIREMENTS

4.7.5.1.1 The ultimate heat sink shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the average water level to be within the limits.

4.7.5.1.2 The isolation valves in the barrier dam between the intake structure and Big Mud Creek shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by cycling each valve through at least one complete cycle of full travel.

Page Intentionally Blank

PLANT SYSTEMS

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The control room emergency ventilation system shall be OPERABLE with:

a. Two booster fans,

b. Two isolation valves in each outside air intake duct,

c. Two isolation valves in the toilet area air exhaust duct,

d. One filter train,

e. At least two air conditioning units, and

f. Two isolation valves in the kitchen area exhaust duct.

NOTE
The control room envelope boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6 or during movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3 and 4:

a. With one booster fan inoperable, restore the inoperable fan 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. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

b. With one isolation valve per air duct inoperable, operation may continue provided the other isolation valve in the same duct is maintained closed; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

c. With the filter train inoperable for reasons other than an inoperable Control Room Envelope boundary:

   1. Immediately initiate action to implement mitigating actions, and

   2. Within 1 hour, verify LCO 3.4.8, "Specific Activity," is met, and

   3. Within 24 hours restore the filter train to OPERABLE status.

   With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With only one air conditioning unit OPERABLE, restore at least two air conditioning units 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. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION: (continued)

MODES 1, 2, 3 and 4: (continued)

NOTE
Action not applicable when second booster fan intentionally made inoperable.

e. With two booster fans inoperable for reasons other than an inoperable Control Room Envelope boundary:

1. Immediately initiate action to implement mitigating actions, and
2. Within 1 hour, verify LCO 3.4.8, "Specific Activity," is met, and
3. Within 24 hours restore at least one booster fan to OPERABLE status.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

NOTE
Action not applicable when third air conditioning unit intentionally made inoperable.

f. With three air conditioning units inoperable for reasons other than an inoperable Control Room Envelope boundary, restore at least one air conditioning unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

g. With the filter train inoperable due to an inoperable Control Room Envelope boundary:

1. Immediately initiate actions to implement mitigating actions, and
2. Within 24 hours, verify mitigating actions to ensure Control Room Envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, and
3. Restore Control Room Envelope boundary to OPERABLE status within 90 days.

With the above requirements not met, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION: (continued)

MODES 5 and 6 or during movement of irradiated fuel assemblies:

a. With one booster fan inoperable, restore the inoperable fan to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency ventilation system in the recirculation mode or suspend movement of irradiated fuel assemblies.

b. With one isolation valve in an air duct inoperable, maintain the other isolation valve in the same air duct closed or suspend movement of irradiated fuel assemblies.

c. With the filter train inoperable, suspend movement of irradiated fuel assemblies.

d. With only one air conditioning unit OPERABLE, restore at least two air conditioning units to OPERABLE status within 7 days or suspend movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.7.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is $\leq$ 120°F.

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

1. Initiating flow through the HEPA filter and charcoal adsorber train and verifying that each booster fan operates for at least 15 minutes.

2. Starting (unless already operating) each air conditioning unit and verifying that it operates for at least 8 hours.

c. By performing required control room emergency ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

SURVEILLANCE REQUIREMENTS (Continued)

d. In accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation signal the system automatically isolates the control room within 35 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

e. By performing required Control Room Envelope unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

PLANT SYSTEMS

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two independent ECCS area exhaust air filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ECCS area exhaust air filter 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 HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each ECCS area exhaust air filter train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

b. By performing required ECCS area ventilation system filter testing in accordance with the Ventilation Filter Testing Program.

c. In accordance with the Surveillance Frequency Control Program:

1. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ASME N510-1989.

2. Verifying that the filter train starts on a Safety Injection Actuation Signal.

Page Deleted

Page Deleted

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

1. Either decontaminated and repaired, or

2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1.1 Test Requirements – Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.9.1.2 Test Frequencies – Each category of sealed sources shall be tested at the frequencies described below.

a. Sources in use (excluding startup sources previously subjected to core flux) – In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive material:

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources - Each sealed startup source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4.7.9.1.3 Reports - A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

PLANT SYSTEMS

3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 All safety related snubbers shall be OPERABLE.

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

ACTION:

With one or more safety related snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Testing Program.

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

DELETED

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

DELETED

TABLE 3.7-2a

DELETED

TABLE 3.7-2a (CONTINUED)

DELETED

TABLE 3.7-2a (CONTINUED)

DELETED

TABLE 3.7-2a (CONTINUED)

DELETED

TABLE 3.7-2b

DELETED

TABLE 3.7-2b (CONTINUED)

DELETED

TABLE 3.7-2b (CONTINUED)

DELETED

TABLE 3.7-2b (CONTINUED)

DELETED

TABLE 3.7-2b (CONTINUED)

DELETED

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 generator sets each with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

2. A separate fuel storage system containing a minimum of 19,000 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action f below:

1. Demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a 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 features(s), declare required features(s) with no offsite power available inoperable when its redundant required features(s) is inoperable.

3. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

4. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION (continued)

NOTE
If the absence of any common-cause failure cannot be confirmed, Surveillance Requirement 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

b. With one diesel generator of 3.8.1.1.b inoperable:

1. Demonstrate the OPERABILITY of the A. C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter.

2. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

3. If the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG.

4. Restore the diesel generator to OPERABLE status within 14 days* or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

5. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

* A one-time AOT extension for the inoperable 1B EDG allows 30 days to restore the EDG to OPERABLE status. Compensatory Measure within FPL Letter L-2019-153 dated July 25, 2019 will remain in effect during the extended AOT period. This extension expires on August 14, 2019 at 0736 hours EDT.

ACTION (continued)

NOTE
1. Enter applicable ACTIONS of LCO 3.8.2.1, "A.C. Distribution - Operating," when ACTION c is entered with no AC power to any train.
2. If the absence of any common-cause failure cannot be confirmed, Surveillance Requirement 4.8.1.1.2.a.4 shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

c. With one offsite A.C. circuit and one diesel generator inoperable:

1. Demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter.

2. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

3. If the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG*.

4. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

5. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

6. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION a or b, as appropriate, with the time requirement of that ACTION based on the time of the initial loss of the remaining inoperable A.C. power source.

d. With two of the 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 one of the inoperable offsite sources to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours.

3. Following restoration of one offsite source, follow ACTION a with the time requirement of that ACTION based on the time of the initial loss of the remaining inoperable offsite A.C. circuit.

ACTION (continued)

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in the at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.

f. With one Unit 1 startup transformer (1A or 1B) inoperable and with a Unit 2 startup transformer (2A or 2B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 2 require the use of the startup transformer administratively available to both units, Unit 1 shall demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

g. LCO 3.0.4.b is not applicable to diesel generators.

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 auxiliary transformer to the startup transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

1. Verifying fuel level in the engine-mounted fuel tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the engine-mounted tank,

SURVEILLANCE REQUIREMENTS (continued)

4. Verifying the diesel starts from ambient condition and accelerates to approximately 900 rpm in less than or equal to 10 seconds**. The generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz within 10 seconds after the start signal**. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual/Local

   b) Simulated loss-of-offsite power by itself.

   c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.

   d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 3500 kW in accordance with the manufacturer's recommendations and operates within a load band of 3300 to 3500 kW*** for at least an additional 60 minutes, and

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

b. By removing accumulated water:

1. From the engine-mounted fuel tank in accordance with the Surveillance Frequency Control Program and after each occasion when the diesel is operated for greater than 1 hour, and

2. From the storage tank in accordance with the Surveillance Frequency Control Program.

** The diesel generator start (10 sec.) from ambient conditions shall be performed in accordance with the Surveillance Frequency Control Program in these surveillance tests. All other diesel generator starts for the purposes of this surveillance testing may be preceded by an engine prelube period and may also include warmup procedures (e.g., gradual acceleration) as recommended by the manufacturer so that mechanical stress and wear on the diesel generator is minimized.

*** The indicated load band is meant as guidance to avoid routine overloading. Variations in loads in excess of the band due to changing bus loads shall not invalidate this test.

SURVEILLANCE REQUIREMENTS (Continued)

c. Verify 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.

d. DELETED

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

1. DELETED

2.

> **NOTE**
>
> Credit may be taken for unplanned events that satisfy this SR.

Verifying generator capability to reject the single largest post-accident load while maintaining voltage at 4160 $\pm$ 420 volts and frequency at 60 $\pm$ 1.2 Hz.

3.

> **NOTE**
>
> This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying that upon an actual or simulated loss of offsite power signal by itself.

a) Deenergization of the emergency busses and load shedding from the emergency busses.

SURVEILLANCE REQUIREMENTS (Continued)

b) The diesel starts on the auto-start signal****, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$ 210 volts and 60 $\pm$ 0.6 Hz during this test.

4.

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying that upon an actual or simiulated ESF actuation signal (without loss-of-offsite power) the diesel generator starts**** on the auto-start signal, and:

a) Within 10 seconds, generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz.

b) Operates on standby for greater than or equal to 5 minutes.

c) Steady-state generator voltage and frequency shall be 4160 $\pm$ 210 volts and 60 $\pm$ 0.6 Hz and shall be maintained throughout this test.

5.

NOTE
This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying that upon an actual or simulated loss-of-offsite power signal in conjunction with an ESF actuation signal:

a) Deenergization of the emergency busses and load shedding from the emergency busses.

**** This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

SURVEILLANCE REQUIREMENTS (Continued)

b) The diesel starts 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 auto-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 ± 210 volts and 60 ± 0.6 Hz during this test.

c) All automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection signal.

6.

NOTE
Credit may be taken for unplanned events that satisfy this SR.

Verifying the diesel generator operates for at least 24 hours****. During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3960 kW# and during the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3300 to 3500 kW#. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

7. DELETED

8.

NOTE
This Surveillance shall not normally be performed in MODE 1, 2, 3 or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon actual or simulated restoration of offsite power.

b) Transfer its load to the offsite power source, and

c) Be restored to its standby status.

This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

**** This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

SURVEILLANCE REQUIREMENTS (continued)

9.

> **NOTE**
> This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying that with the diesel generator operating in a test mode (connected to its bus), an actual or simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

10. DELETED

11.

> **NOTE**
> This Surveillance shall not normally be performed in MODE 1 or 2. However, the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines that the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

Verifying that the automatic load sequence timers are operable with the interval between each load block within + 1 second of its design interval.

f. In accordance with the Surveillance Frequency Control Program or after any modification which could affect diesel generator independence by starting**** the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to approximately 900 rpm in less than or equal to 10 seconds.

g. In accordance with the Surveillance Frequency Control Program by performing a pressure test of those portions of the diesel fuel oil system designed to USAS B31.7 Class 3 requirements in accordance with the Inservice Inspection Program.

4.8.1.1.3 <u>Reports</u> – (Not Used)

4.8.1.1.4 The Class 1E underground cable system shall be demonstrated OPERABLE within 30 days after the movement of any loads in excess of 80% of the ground surface design basis load over the cable ducts by pulling a mandrel with a diameter of at least 80% of the duct's inside diameter through a duct exposed to the maximum loading (duct nearest the ground's surface) and verifying that the duct has not been damaged.

This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

**** This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

(NOT USED)

ELECTRICAL POWER SYSTEMS

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 set with:

1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

2. A fuel storage system containing a minimum of 19,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

NOTE
Enter the ACTION of LCO 3.8.2.2, "A.C. 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, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. 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 top of irradiated fuel assemblies seated within the reactor vessel, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2.1 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2a.5.

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generator sets:

4160	volt Emergency Bus	1A3
4160	volt Emergency Bus	1B3
480	volt Emergency Bus	1A2
480	volt Emergency Bus	1B2
480	volt Emergency MCC Busses	1A5, 1A6, 1A7
480	volt Emergency MCC Busses	1B5, 1B6, 1B7
120	volt A.C. Instrument Bus	1MA
120	volt A.C. Instrument Bus	1MB
120	volt A.C. Instrument Bus	1MC
120	volt A.C. Instrument Bus	1MD

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

> **NOTE**
>
> Enter applicable ACTIONS of LCO 3.8.2.3, "D.C. Distribution - Operating," for DC trains made inoperable by inoperable AC distribution system.

a. With less than the above complement of A.C. emergency busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Instrument Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. Instrument Bus within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) re-energize the A.C. Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators in accordance with the Surveillance Frequency Control Program by verifying indicated power availability.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Bus

3 - 480 volt Emergency MCC Busses

2 - 120 volt A.C. Instrument Busses

APPLICABILITY: MODES 5 and 6

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators in accordance with the Surveillance Frequency Control Program by verifying indicated power availability.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt D.C. bus No. 1A, 125-volt Battery bank No. 1A and a full capacity charger.

b. 125-volt D.C. bus No. 1B, 125-volt Battery bank No. 1B and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks or busses inoperable, restore the inoperable battery bank or bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.3.2.a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying indicated power availability.

4.8.2.3.2 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 129-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:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, and

3. The average electrolyte temperature of 10% (60 cells total) of connected cells is above 50°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 anti-corrosion material,

3. Battery cell inter-connection resistance values are maintained at the values below:

Battery Inter-Connection Measurement Limits		
Battery Inter-Connection Type	Maximum Individual Inter-Connection Resistance	Maximum Average Inter-Connection Resistance [Battery Bank*]
Inter-Cell	≤ 150 x 10-6 ohms	≤ 50 x 10-6 ohms
Inter-Tier	≤ 200 x 10-6 ohms	
Inter-Rack	≤ 200 x 10-6 ohms	
Output Terminal	≤ 150 x 10-6 ohms	

* The battery bank average interconnection resistance limit is the average of all inter-cell, inter-tier, inter-rack and output terminal connection resistance measurements for all series connections in the battery string

and,

4. The battery charger will supply at least 300 amperes at 140 volts for at least 6 hours.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.3.2.d.

f. Annual performance discharge tests of battery capacity shall be given 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 REQUIREMENT

	CATEGORY A[1]	CATEGORY B[2]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	>Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	≥ 1.195[b]	≥ 1.190 Average of all connected cells > 1.200	Not more than .020 below the average of all connected cells Average of all connected cells ≥ 1.190[b]

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps when on charge.

(c) Corrected for average electrolyte temperature.

(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) With any Category B parameter not within its allowable value, declare the battery inoperable.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 125-volt D.C. bus, and

1 - 125-volt battery bank and charger supplying the above D.C. bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying indicated power availability.

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

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling cavity shall be maintained within 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 at $\geq$ 40 gpm of greater than or equal to 1900 ppm boron or its equivalent to restore boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration limit shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the refueling cavity shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

---

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two wide range logarithmic neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each wide range logarithmic neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the start of CORE ALTERATIONS, and

c. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for a minimum of 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

REFUELING OPERATIONS

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment 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.

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by isolation valve, blind flange, or manual valve except for valves that are open on an intermittent basis under administrative control, or

2. Be capable of being closed by an OPERABLE automatic containment isolation valve, or

3. Be capable of being closed by an OPERABLE containment vacuum relief valve.

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of 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. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment isolation valve within 72 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel in the containment by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing of containment isolation valves per the applicable portions of Specifications 4.6.3.1.1. and 4.6.3.1.2.

INTENTIONALLY DELETED

INTENTIONALLY DELETED

INTENTIONALLY DELETED

REFUELING OPERATIONS

SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be OPERABLE and in operation*.

APPLICABILITY: MODE 6 when the water level above the top of irradiated fuel assemblies seated within the reactor pressure vessel is greater than or equal to 23 feet.

ACTION:

a. With less than one shutdown cooling loop in operation, suspend all operations involving an increase in reactor decay heat load or operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm in accordance with the Surveillance Frequency Control Program.

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

* The shutdown cooling 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 reactor pressure vessel hot legs, provided no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.9.1.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, within one (1) hour 1) initiate corrective action to return the required loops to OPERABLE status, or 2) establish greater than or equal to 23 feet of water above irradiated fuel assemblies seated within the reactor pressure vessel.

b. With no shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1. and within one (1) hour initiate corrective action to return the required shutdown cooling loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm in accordance with the Surveillance Frequency Control Program.

4.9.8.2.1 Verify shutdown cooling loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

---

* One required shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing, provided that the other shutdown cooling loop is OPERABLE and in operation.

REFUELING OPERATIONS

CONTAINMENT ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment isolation system shall be OPERABLE.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTION:

With the containment isolation system inoperable, either suspend all operations involving movement of recently irradiated fuel assemblies within containment or close each of the penetrations providing direct access from the containment atmosphere to the outside atmosphere.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel assemblies by verifying that containment isolation occurs on manual initiation and on a high radiation signal from two of the containment radiation monitoring instrumentation channels.

REFUELING OPERATIONS

WATER LEVEL – REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During CORE ALTERATIONS.
During movement of irradiated fuel assemblies within containment.

ACTION:

With the requirements of the above specifications not satisfied, immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within containment, and immediately initiate action to restore refueling cavity water level to within limits.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

REFUELING OPERATIONS

SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The Spent Fuel Pool shall be maintained with:

a. The fuel storage pool water level greater than or equal to 23 ft over the top of irradiated fuel assemblies seated in the storage racks, and

b. The fuel storage pool boron concentration greater than or equal to 1900 ppm.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

a. With the water level requirement not satisfied, immediately 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. With the boron concentration requirement not satisfied, immediately suspend all movement of fuel assemblies in the fuel storage pool and initiate action to restore the fuel storage pool boron concentration to within the required limit.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel 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.

4.9.11.1 Verify the fuel storage pool boron concentration is within limit in accordance with the Surveillance Frequency Control Program.

REFUELING OPERATIONS

FUEL POOL VENTILATION SYSTEM – FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12 At least one fuel pool ventilation system shall be OPERABLE.

APPLICABILITY: Whenever recently irradiated fuel is in the spent fuel pool.

ACTION:

a. With no fuel pool ventilation system OPERABLE, suspend all operations involving movement of recently irradiated fuel within the spent fuel pool or crane operation with loads over the recently irradiated spent fuel until at least one fuel pool ventilation system is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel pool ventilation system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train 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:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ≥ 85% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989 (30°C, 95% RH). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 10,350 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. 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 < 4.15 inches Water Gauge while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that the ventilation system maintains the spent fuel storage pool area at a negative pressure of ≥ 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

INTENTIONALLY DELETED

DELETED

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 CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 40 gpm of 1900 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 40 gpm of 1900 ppm boric acid solution 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 CEA required either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended.

SPECIAL TEST EXCEPTIONS

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 This specification deleted.

SURVEILLANCE REQUIREMENTS

4.10.3 This specification deleted.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 This specification deleted.

SURVEILLANCE REQUIREMENTS

4.10.4 This specification deleted.

SPECIAL TEST EXCEPTIONS

CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.5 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.5.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.5.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

Pages 3/4 11-2 through 3/4 11-13 (Amendment No. 123) have been deleted from the Technical Specifications. The next page is 3/4 11-14.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 The concentration of oxygen in the waste gas decay tank shall be determined to be within the above limits by continuously* monitoring the waste gases in the on service waste gas decay tank.

4.11.2.5.2 With the oxygen concentration in the on service waste gas decay tank greater than 2% by volume as determined by Specification 4.11.2.5.1, the concentration of hydrogen in the waste gas decay tank shall be determined to be within the above limits by gas partitioner sample at least once per 24 hours.

---

* When continuous monitoring capability is inoperable, waste gases shall be monitored in accordance with the actions specified for the Waste Gas Decay Tanks Explosive Gas Monitoring System in Chapter 13 of the Updated Final Safety Analysis Report.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 165,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank when reactor coolant system activity exceeds 518.9 μCi/gram DOSE EQUIVALENT XE-133.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE LOCATION

The St. Lucie nuclear units are located on Hutchinson Island in St. Lucie County, about halfway between the cities of Fort Pierce and Stuart on the east coast of Florida. The radius of the exclusion area is 0.97 miles from the center of the St. Lucie Plant. The low population zone (LPZ) includes that area within one mile of the center of the St. Lucie Plant.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel, having the shape of a right circular cylinder with a hemispherical dome and ellipsoidal bottom, surrounded by a reinforced concrete shield building. The radius of the shield building is at least 4 feet greater than the radius of circular cylinder portion of the containment vessel at any point.

5.2.1.1 CONTAINMENT VESSEL

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 232 feet.

c. Net free volume = 2.5×10^6 cubic feet.

d. Nominal thickness of vessel walls = 2 inches.

e. Nominal thickness of vessel dome = 1 inch.

f. Nominal thickness of vessel bottom = 2 inches.

Figure 5.1-1 Deleted

FIGURE 5.1-2

(Deleted)

2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet

b. Annulus nominal volume = 543,000 cubic feet

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 230.5 feet

d. Nominal inside diameter = 148 feet

e. Cylinder wall minimum thickness = 3 feet

f. Dome minimum thickness = 2.5 feet

g. Dome inside radius = 112 feet

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment vessel is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

PENETRATIONS

5.2.3 Penetrations through the containment structure are designed and shall be maintained in accordance with the original design provisions contained in Sections 3.8.2.1.10 and 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or M5 clad fuel rods and/or poison rods, with fuel rods having 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 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 non-limiting core regions.

5.3.1.1 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and no part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the original design provisions contained in Section 4.2.3.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

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 700°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.a The spent fuel pool and spent fuel storage racks shall be maintained with:

1. k_{eff} less than 1.0 when fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

2. A nominal 10.12 inches center to center distance between fuel assemblies in Region 1 of the spent fuel pool storage racks, a nominal 10.30 inches center to center distance between fuel assemblies in the Region 1 cask pit storage rack, and a nominal 8.86 inches center to center distance between fuel assemblies in Region 2 of the spent fuel pool storage racks.

3. A k_{eff} less than or equal to 0.95 when flooded with water containing 500 ppm boron, including an allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

CRITICALITY (Continued)

4. For storage of enriched fuel assemblies, requirements of Criteria in 5.6.1.a.1 and 5.6.1.a.3 shall be met by positioning fuel in the spent fuel storage racks consistent with the requirements of Specification 5.6.1.c.

5. Vessel Flux Reduction Assemblies (VFRAs), as defined in Section 9.1 of the Updated Final Safety Analysis Report, may be placed in any allowable fuel storage location.

6. Fissile material, not contained in a fuel assembly lattice, shall be stored in accordance with the requirements of Criteria in 5.6.1.a.1 and 5.6.1.a.3.

7. The Metamic neutron absorber inserts shall have a ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$.

b. The Region 1 cask pit storage rack shall contain neutron absorbing material (Boral) between stored fuel assemblies when installed in the spent fuel pool.

c. Loading of spent fuel storage racks shall be controlled as described below. Criteria in 5.6.1.c.2, 5.6.1.c.3, 5.6.1.c.5 and 5.6.1.c.6 do not apply to the Region 1 cask pit storage rack.

1. The maximum initial planar average U-235 enrichment of any fuel assembly inserted in a spent fuel storage rack shall be less than or equal to 4.6 weight percent.

2. Fuel placed in Region 1 of the spent fuel pool storage racks shall comply with the storage patterns and alignment restrictions of Figure 5.6-1 and the minimum burnup requirements of Table 5.6-1.

3. Fuel placed in Region 2 of the spent fuel pool storage racks shall comply with the storage patterns or allowed special arrangements of Figure 5.6-2 and the minimum burnup requirements of Table 5.6-1. The allowed special arrangement for fresh fuel may be repeated, provided the applicable interface requirements specified by the safety analysis are met.

4. Any fuel satisfying criteria 5.6.1.c.1, including fresh fuel, may be placed in the Region 1 cask pit storage rack.

5. The same directional orientation for Metamic inserts is required for contiguous groups of 2x2 arrays where Metamic inserts are required.

6. Any 2x2 array of Region 2 storage cells that interface with Region 1 shall comply with the rules of Figure 5.6-3. The allowed special arrangement in Region 2 as shown in Figure 5.6-2 shall not be placed adjacent to Region 1.

d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a maximum planar average U-235 enrichment less than or equal to 4.6 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

DRAINAGE

5.6.2 The fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

5.6.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1706 fuel assemblies, and the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 143 fuel assemblies. The total Unit 1 spent fuel pool and cask pit storage capacity is limited to no more than 1849 fuel assemblies.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as seismic Class I in Section 3.2.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 3.7 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirement.

5.8 DELETED

5.9 DELETED

Allowable Checkerboard Storage Patterns
(See Notes 4 and 5)

Pattern "A"
See Note 1

F	WH
WH	F

Pattern "B"
See Note 2

1	1
1	1

Pattern "C"
See Note 3

2	2
WH	2

NOTES:

1. F represents Fresh Fuel. WH represents an empty cell. Allowable Pattern is Fresh Fuel checkerboarded with empty cells. Diagram is for illustration only.

2. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 1 is defined in Table 5.6-1. Allowable pattern is at least one insert [either Metamic or full-length full-strength CEA] in any one of the 2x2 array locations. Diagram is for illustration only.

3. Numbering denotes fuel assembly type. WH represents an empty cell. Minimum burnup for fuel assembly type 2 is defined in Table 5.6-1. Allowable pattern is at least one empty cell in any of the 2x2 array locations. Diagram is for illustration only.

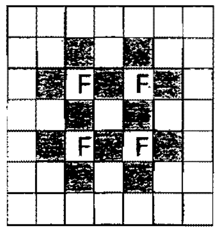
4. The storage arrangements of fuel within a rack module may contain more than one pattern. Each cell is a part of up to four 2x2 arrays, and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

5. Empty cells within any pattern are acceptable.

FIGURE 5.6-1
Allowable Region 1 Storage Patterns and Fuel Arrangements

ALLOWED SPECIAL ARRANGEMENT

Fresh Fuel Assemblies in Pattern "C", "D", or "E" Racks



F = FRESH FUEL ASSEMBLY

= EMPTY CELL

ALLOWABLE CHECKERBOARD STORAGE PATTERNS (See Notes 4 and 5)

Pattern "C" See Note 1	Pattern "D" See Note 2	Pattern "E" See Note 3

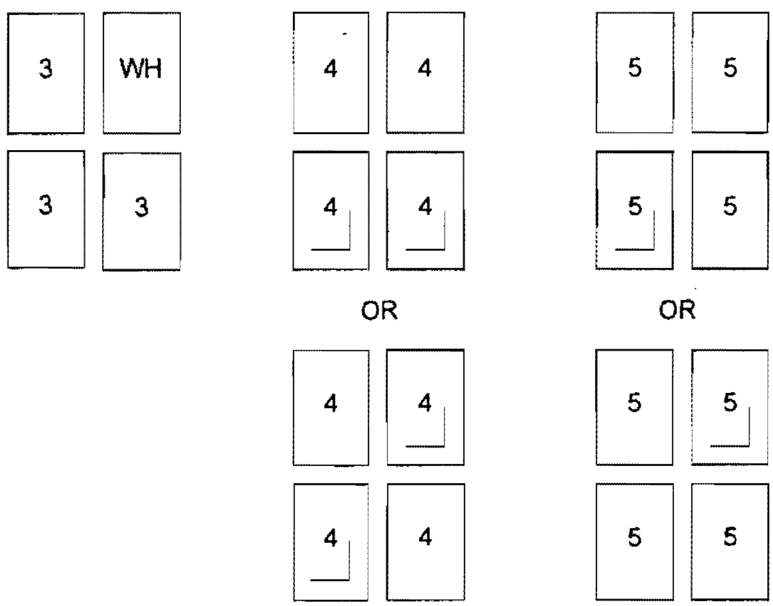


FIGURE 5.6-2 (Sheet 1 of 2)
Allowable Region 2 Storage Patterns and Fuel Alignments

NOTES to Figure 5.6-2

NOTES:

1. Numbering denotes fuel assembly type. WH represents an empty cell. Minimum burnup for fuel assembly type 3 is defined in Table 5.6-1. Allowable pattern is at least one empty cell in any of the 2x2 array locations. Diagram is for illustration only.
2. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 4 is defined in Table 5.6-1. Allowable pattern is at least two inserts, (either Metamic or full-length, full-strength CEA) in the 2x2 array. Diagrams are for illustration only.
3. Numbering denotes fuel assembly type. Minimum burnup for fuel assembly type 5 is defined in Table 5.6-1. Allowable pattern is one insert, (either Metamic or full-length, full-strength CEA) in the 2x2 array. Diagrams are for illustration only.
4. The storage arrangements of fuel within a rack module may contain more than one pattern. 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.
5. Empty cells within any pattern are acceptable.

FIGURE 5.6-2 (Sheet 2 of 2)
Allowable Region 2 Storage Patterns and Fuel Arrangements

Allowed Region 2 to Region 1 Fuel Alignments
(See Notes 4 and 5)

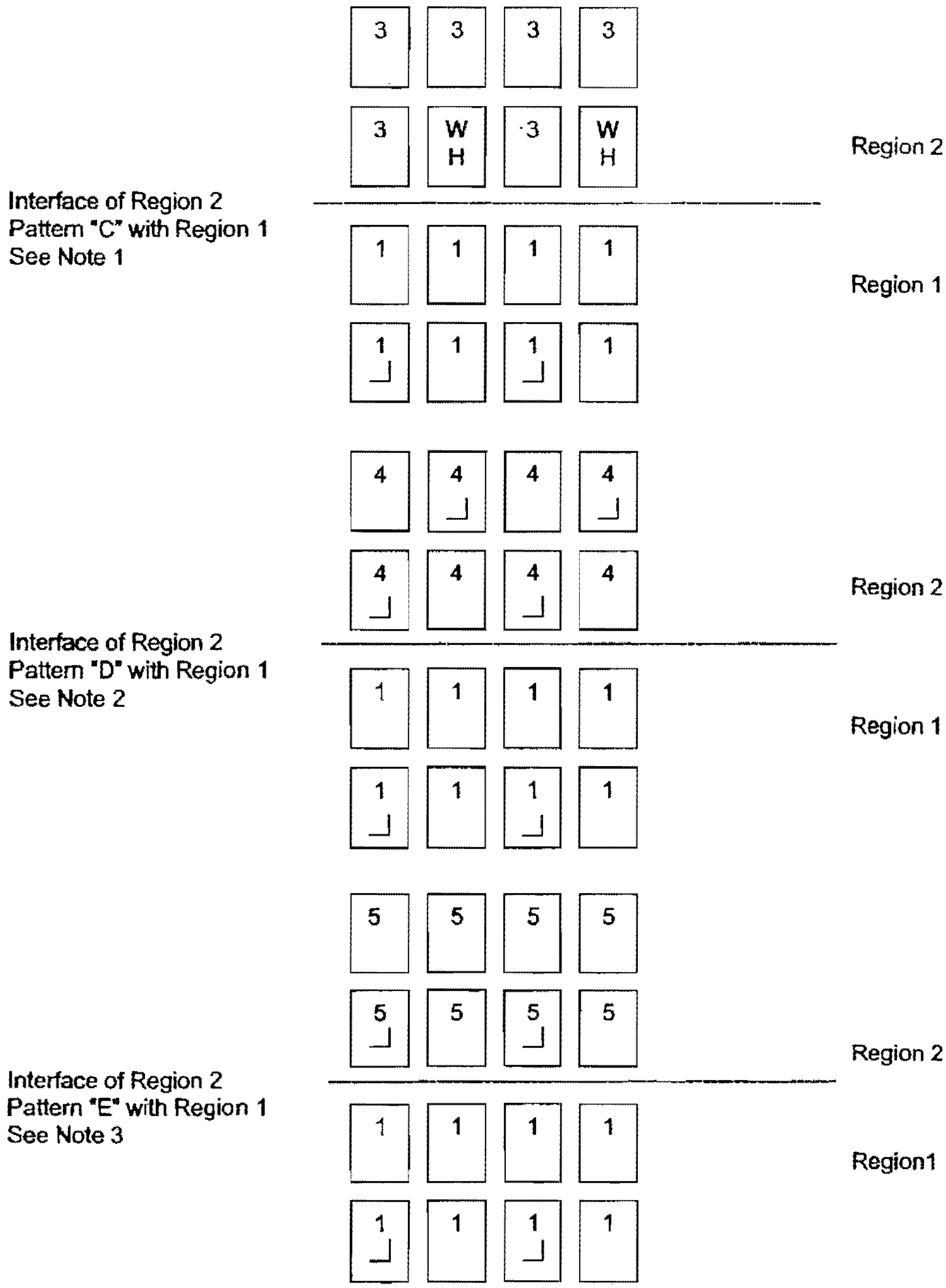


FIGURE 5.6-3 (Sheet 1 of 2)
Region 2 Interface Requirements with Region 1

NOTES to Figure 5.6-3

NOTES:

1. WH represents an empty cell. For the interface of Pattern "C" with Region 1, the empty cell must be on the rack periphery facing Region 1 racks. Diagrams are for illustration only.
2. For the interface of pattern "D" with Region 1, at least one cell on the rack periphery facing Region 1 rack must contain an insert (either Metamic of full-length full-strength CEA) in the 2x2 array. If the insert is Metamic, the insert must be oriented so that the corner of the L-shape is located closest to the Region 1 rack. Diagram is for illustration only.
3. For the interface of Pattern "E" with Region 1, the insert must be on the rack periphery facing the Region 1 rack. The insert may be either a Metamic of full-length full strength CEA. If the insert is Metamic, the insert must be oriented so that the corner of the L-shape is located closest to the Region 1 rack. Diagram is for illustration only.
4. Empty cells with any pattern are acceptable.
5. There are no interface requirements within Region 1. Any Pattern within Region 1 may be used for the interface. Pattern "B" was used only as an illustration.

FIGURE 5.6-3 (Sheet 2 of 2)
Region 2 Interface requirements with Region 1

TABLE 5.6-1
Minimum Burnup as a Function of Enrichment

Fuel Type	Cooling Time (Years)	Coefficients		
		A	B	C
1	0	-36.6860	22.4942	-1.4413
2	0	-36.1742	16.6000	-0.8958
3	0	-34.7091	23.1361	-1.6204
4	0	-24.5145	21.3404	-1.2444
	2.5	-26.8311	22.5246	-1.5029
	5	-24.7233	20.9763	-1.3246
	10	-23.6285	19.9541	-1.2505
	15	-23.5458	19.9336	-1.3180
	20	-22.4382	19.2460	-1.2629
5	0	-8.1856	14.5275	-0.0719
	2.5	-11.8506	16.1475	-0.3969
	5	-16.5196	18.5309	-0.7837
	10	-13.6831	16.3475	-0.5844
	15	-12.5819	15.6175	-0.5656
	20	-12.6469	15.4575	-0.5906

NOTES:

1. To qualify in a "fuel type," the burnup of a fuel assembly must exceed the minimum burnup "BU" calculated by inserting the "coefficients" for the associated "fuel type" and "cooling time" into the polynomial function:

   $BU = A + B*E + C*E^2$, where:

   BU = Minimum Burnup (GWD/MTU)

   E = Initial Maximum Planar Average Enrichment (weight percent uranium-235)

   A, B, C = Coefficients

2. Interpolation between values of cooling time is not permitted.

DELETED

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor shall be responsible for the control room command function. During any absence of the Shift Supervisor from the control room while either 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 Shift Supervisor from the control room while both units are 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

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. This onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Quality Assurance Topical Report and updated in accordance with 10 CFR 50.54(a)(3). The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR or the Quality Assurance Topical Report.

b. A specified corporate officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The plant manager shall be responsible for overall safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the radiation protection manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

6.2 ORGANIZATION (continued)

UNIT STAFF

6.2.2 The unit staff organization shall meet the requirements of 10 CFR 50.54(m) and include the following:

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. A minimum of three non-licensed operators is required when both units are in MODES 5 or 6.

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.e 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 health physics 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. The operations manager or assistant operations manager shall hold an SRO license.

e. 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, or 4 by use of either a dedicated STA, a Shift Supervisor who meets the qualifications for the STA as required by Technical Specification 6.3.1, or an individual assigned to the unit with a Senior Reactor Operator's license who meets the qualifications for the STA as required by Technical Specification 6.3.1. If the STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the individual may fill the same position on Unit 2.

DELETED

DELETED

DELETED

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for:

(1) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975,

(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:

- Bachelor's degree in engineering from an accredited institution; or
- Professional Engineer's (PE) license obtained by successful completion of the PE examination; or
- Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or
- Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

6.3.2 For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.4 DELETED

6.5 DELETED

DELETED

DELETED

DELETED

DELETED

DELETED

6.6 DELETED

6.7 DELETED

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirements of NUREG 0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Not Used.

e. Not Used.

f. Not Used.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Control Program for effluent monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974.

j. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 DELETED

6.8.3 DELETED

6.8.4 The following programs shall be established, implemented, and maintained.

a. Primary Coolant Sources Outside Containment

A program to reduce leakage 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 Shutdown Cooling System, High Pressure Safety Injection System, Containment Spray System, and RCS Sampling. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at least once per 18 months.

The provisions of Specification 4.0.2 are applicable.

b. In-Plant Radioiodine Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) 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:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions, and

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

d. Backup Method for Determining Subcooling Margin

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

(i) Training of personnel, and

(ii) Procedures for monitoring.

e. DELETED

f. 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 10 CFR 20.1001 - 20.2401, Appendix B, Table 2, Column 2.

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 on a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative 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 contributions 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,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the SITE BOUNDARY shall be limited to the following:

   a) For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

   b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ;

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radio-nuclides 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 the environmental exposure pathways. The program shall (1) be contained in the ODCM,

(2) conform to the guidance of Appendix I to 10 CFR Part 50, and
(3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program 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 is in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," except that the next Type A test performed after the December 8, 2005 Type A test shall be performed no later than December 8, 2020.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 42.8 psig. The containment design pressure is 44 psig.

The maximum allowed containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Leakage rate acceptance criteria are:

a. 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 $< 0.60\ L_a$ for the Type B and C tests, $\leq 0.75\ L_a$ for Type A tests, and $\leq 0.096\ L_a$ for secondary containment bypass leakage paths.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For the personnel air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq 1.0\ P_a$.

3) For the emergency air lock door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq$ 10 psig.

The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.

The provisions of T.S. 4.0.3 are applicable to the Containment Leak Rate Testing Program.

i. Deleted

j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a. a change in the TS incorporated in the license; or

   b. a change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.j.2.a or 6.8.4.j.2.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. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 3.

1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass less than the value specified below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Penetration	Flowrate
Control Room Emergency Ventilation	$\leq$ 0.05%	2000 $\pm$ 200 cfm
Shield Building Ventilation System	$\leq$ 0.05%	6000 $\pm$ 600 cfm
ECCS Area Ventilation System	$\leq$ 0.05%	30,000 $\pm$ 3000 cfm

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Penetration	Flowrate
Control Room Emergency Ventilation	$\leq$ 0.05%	2000 $\pm$ 200 cfm
Shield Building Ventilation System	$\leq$ 0.05%	6000 $\pm$ 600 cfm
ECCS Area Ventilation System	$\leq$ 0.05%	30,000 $\pm$ 3000 cfm

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Emergency Ventilation	$\leq$ 2.5%	70%
Shield Building Ventilation System	$\leq$ 2.5%	70%
ECCS Area Ventilation System	$\leq$ 2.5%	70%

4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency Ventilation	< 4.15" W.G.	2000 $\pm$ 200 cfm
Shield Building Ventilation System	$\leq$ 6.15" W.G.	6000 $\pm$ 600 cfm
ECCS Area Ventilation System	< 4.15" W.G.	30,000 $\pm$ 3000 cfm

5. At least once per 18 months, demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	Wattage
Shield Building Ventilation System	
Main Heaters	30 $\pm$ 3 kW
Auxiliary Heaters	1.5 $\pm$ 0.25 kW

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

l. Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and 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 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 0.5 gpm total through all SGs and 0.25 gpm through any one SG.

   3. The operational leakage performance criterion is specified in LCO 3.4.6.2.c, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube repair 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.

l. Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair 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. An assessment of degradation shall be performed to determine the type and location of flaws to which the tube 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 replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). 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.

m. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

m. Control Room Envelope Habitability Program (continued)

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, 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.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

n. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(i) 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 or a water and sediment content within limits;

(ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

(iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

o. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of frequencies of those Surveillance Requirements for which the frequency is controlled by the program.

b. Changes to the frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the frequencies established in the Surveillance Frequency Control Program.

p. Snubber Testing Program

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

1. This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.

2. The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to the conditions listed in 10 CFR 50.55a(b) and subject to Commission approval.

3. The program shall, as required by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements" and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants".

4. The 120-month program updates shall be made in accordance with 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(3)(v) and 10 CFR 50.55a(b) (including 10 CFR 50.55a(b)(3)(v)) subject to the conditions listed therein.

q. Component Cyclic or Transient Limit Program

The program provides controls to track the FSAR, Section 5.2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

r. Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b: Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, November 2006. The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 1 and 2;

c. When a RICT is being used, any plant configuration change within the scope of the Configuration Risk Management Program must be considered for the effect on the RICT.

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required If the plant configuration change would lower plant risk and would result in a longer RICT.

d. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

e. If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

1. Numerically accounting for the increased possibility of CCF in the RICT calculation, or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

s. 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 inoperabilites, 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 supported system is also inoperable, or

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.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following:

(1) receipt of an operating license,

(2) amendment of the license involving a planned increase in power level,

(3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and

(4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS 1/

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Annual reports shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS

6.9.1.6 Deleted

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.7 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operation shall be submitted within 60 days after January 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.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

6.9.1.8 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

**A single submittal may be made for a multiple unit station.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.11 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater Than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Full Length CEA Position – Misalignment > 15 inches
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factor – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the following documents, approved Revisions and Supplements as specified in the COLR.

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A) [also issued as XN-NF-75-27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors"

4. DELETED

5. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations"

CORE OPERATING LIMITS REPORT (continued)

6. DELETED

7. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"

8. DELETED

9. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors"

10. XN-NF-621(P)(A), " Exxon Nuclear DNB Correlation for PWR Fuel Designs"

11. DELETED

12. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"

13. ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU"

14. XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

15. DELETED

16. DELETED

17. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Design"

18. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"

19. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results"

CORE OPERATING LIMITS REPORT (continued)

20. EMF-1961 (P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors."

21. EMF-231O(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Revision 1, as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0.

22. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," Revision 0, as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0.

23. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Revision 0, as supplemented by ANP-2903(P), "St. Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA summary Report with Zr-4 Fuel Cladding," Revision 1.

24. BAW-10240(P)(A) Revision 0, "Incorporation of M5 Properties in Framatome ANP Approved Methods."

25. WCAP-16045-P-A, Revision 0, “Qualification of the Two-Dimensionall Transport Code PARAGON,” August 2004.

CORE OPERATING LIMITS REPORT (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.12 A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.l, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) 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)*. 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 which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which 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 level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., 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 by the facility Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after the approval of the plant manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

1. Shall be documented and this documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 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.

2. Shall become effective after the approval of the plant manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

DELETED

<u>APPENDIX B - PART I</u>

DELETED

APPENDIX B - PART II

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

TECHNICAL SPECIFICATIONS

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE UNIT NO. 1

OPERATING LICENSE NO. DPR-67

Docket No. 50-335

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of the local area environment of the St. Lucie Nuclear Plant during construction and operation.

The principle objectives of the EPP are to:

1. Verify that the plant is operated in an environmentally acceptable manner as established by the FES and other NRC environmental impact assessments

2. Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection

3. Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects

Environmental concerns identified in the Unit 1 FES which relate to water quality matters are to be regulated by way of the licensee's Wastewater permit

2.0 Environmental Protection Issues

In the FES-OL dated June 1973, NRC staff considered the environmental impacts associated with the operation of the St. Lucie Plant Unit 1. Certain environmental issues were identified which required study or license conditions for resolution of environmental concerns and to assure adequate environmental protection. The Unit 1 Appendix B Environmental Technical Specifications accompanying license DPR-67 included discharge restrictions and monitoring programs to resolve the issues. Prior to issuance of this EPP, ETS requirements related to non-radiological environmental activities have included the following programs:

2.1 Aquatic monitoring programs to insure:

1. Protection of the local aquatic communities by limiting thermal stress to aquatic organisms

2. Minimization of cooling system organism entrainment and impingement levels

3. Protection of local aquatic biota by minimizing the release of chlorine used to control cooling system biofouling to that necessary to maintain plant efficiency and integrity

4. That the local aquatic environment is protected from potential discharges of heavy metals, discharge of water with unacceptable pH from the plant and insuring that no significant dissolved oxygen alteration due to plant operation occurred

To insure that the issues identified in items 1, 2, 3 and 4 above have and are being satisfied, extensive chemical, thermal and biotic monitoring has been performed since plant operation began in 1976.

With assumption of aquatic monitoring programs by EPA through the NPDES program, as delineated in NPDES Permit FL0002208 effective January 29, 1982, NRC will rely on EPA for resolution of issues involving the monitoring of water quality and aquatic biota.

On May 1, 1995, the FDEP was granted authority by the U.S. Environmental Protection Agency (EPA) to administer the NPDES permitting programs. Pursuant to the Florida Administrative Code (FAC) 62-620.105(10), the EPA-issued NPDES permit and the State-issued wastewater permit for each facility were to be combined into one document. The resulting single document, Wastewater Permit No. FL0002208, combines the NPDES Permit FL0002208 and the State Wastewater Permit IO56-194945.

2.2 Terrestrial issues raised have led to programs on sea turtles that:

1. Document the nesting at the site and vicinity; determine effects of the discharge thermal plume on nesting patterns and hatchling migration; and investigate thermal stress on hatching and rearing factors by using turtle eggs from displaced nests

2. Minimize turtle hatchling disorientation by planting a light screen along the beach

The above programs specifically addressed as conditions in the Unit 1 FES, Operating License and Technical Specifications have been completed and the requirements have been satisfied.

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 changes, tests or experiments do not involve an unreviewed environmental question. Changes in plant design or operation or performance of tests or experiments which do not affect the environment are not subject to this requirement.

Before engaging in unauthorized construction or operational activities which may affect the environment, the licensee shall perform an environmental evaluation of such activity.* When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activities and obtain prior approval from the NRC.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the final environmental statement (FES) as modified by staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level (in accordance with 10 CFR Part 51.5(b)(2) 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 a written evaluation which provides bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question.

*Activities are excluded from this requirement if all measurable nonradiological effects are confined to the on-site areas previously disturbed during site preparation and plant construction.

Activities governed by Section 3.3 of this EPP are not subject to the requirements of this section.

3.2 Reporting related to the Wastewater Permit and State Certification (pursuant to Section 401 of the Clean Water Act)

1. Violations of the Wastewater Permit or the State 401 Certification Conditions shall be reported to the NRC by submittal of copies of the reports required by the Wastewater Permit or State 401 Certification.

2. The licensee shall provide the NRC with a copy of any 316(b) studies and/or related documentation at the same time it is submitted to the permitting agency.

3. Changes and additions to the Wastewater Permit or the State 401 Certification shall be reported to the NRC within 30 days following the date the change 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.

4. The NRC shall be notified of changes to the effective Wastewater 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 Wastewater 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 or approval from other Federal, State, or 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 the indicates or could result in significant environmental impact causally related to station operation shall be recorded and promptly reported to the NRC Operations Center within 72 hours via Emergency Notification System described in 10 CFR 50.72. In addition, the reporting requirement time frame shall be consistent with 10 CFR 50.72 for environmental protection issues. The initial report shall be followed by a written report as described in Section 5.4.2. No routine monitoring programs are required to implement this condition. Events covered by Section 3.2 of this EPP will be subject to reporting requirements as defined in that section and not subject to these requirements.

The following are examples of unusual or important events: excessive bird impaction events; onsite plant or animal disease outbreaks; mortality (causally related to station operation), or unusual occurrence of any species protected by the Endangered Species Act of 1973; unusual fish kills; increase in nuisance organisms or conditions; and unanticipated or emergency discharge of waste water or chemical substances.

4.2 Terrestrial/Aquatic Issues

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and indirectly, aquatic biota. The NRC will rely on the decisions made by the State of Florida under the authority of the Clean Water Act and, in the case of sea turtles, decisions made by the NMFS under the authority of the Endangered Species Act, for any requirements pertaining to terrestrial and aquatic monitoring.

In accordance with Section 7(a) of the Endangered Species Act, the NMFS issued a Biological Opinion that prescribes an Incidental Take Statement (ITS) and mandatory terms and conditions. The currently applicable Biological Opinion concludes that continued operation of the St. Lucie Plant circulating seawater cooling system is not likely to jeopardize the continued existence of the listed species or to destroy or adversely modify the designated critical habitat of the loggerhead sea turtle.

FPL shall adhere to the specific requirements within the ITS in the currently applicable Biological Opinion. Changes to the ITS or the terms and conditions must be preceded by consultation between the NRG, as the authorizing agency, and NMFS.

4.2.1 DELETED

4.2.2 DELETED

INTENTIONALLY DELETED

INTENTIONALLY DELETED

4.2.3 Light Screen to Minimize Turtle Disorientation

Suitable plants (i.e., native vegetation such as live oak, native figs, wild tamarind, and others) shall be planted and maintained as a light screen along the beach dune line bordering the plant property to minimize turtle disorientation. In addition, FPL owner controlled area lighting shall be shielded so that none of the light is diverted skyward.

4.3 General Exceptions

The environmental conditions of the EPP Section 4 are contingent upon licensee or its contractors being able to obtain the necessary FDEP endangered species permits to take, handle, and experiment with sea turtles. If licensee is unable to obtain the necessary permits, then NRC shall be notified of alternatives by the licensee.

5.0 Administrative Procedures

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the Environmental Protection Plan. 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 plant 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 plant structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the plant. 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

Request for change in the Environmental Protection Plan 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 Environmental Protection Plan.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

5.4.1.1Monthly Reports

Copies of monthly reports covering sea turtle entrapment, capture, rehabilitation, and sea turtle mortalities shall be furnished to NMFS.

5.4.1.2Annual Environmental Operating Report

An Annual Environmental Operating Report describing implementation of this EPP for the previous calendar year shall be submitted to the NRC prior to May 1 of each year.

The report shall include summaries and analyses of the results of the environmental protection activities required by Section 4.2 of this Environmental Protection Plan for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous non-radiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends towards irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of action to alleviate the problem.

The Annual Environmental Operating Report shall also include:

(a) A list of EPP noncompliances and the corrective actions taken to remedy them.

(b) 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 issue.

(c) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

(d) A discussion of the sea turtle entrapment, capture efforts, turtle mortalities, available information on barrier net inspections and maintenance, and the Taprogge condenser tube cleaning system operation including sponge ball loss at St. Lucie Plant

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 data 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 in accordance with 10 CFR 50.4 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 a copy of such reports within 30 days of the date they submitted to the other agency.