UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20355

SOUTH CAROLINA ELECTRIC & GAS COMPANY
SOUTH CAROLINA PUBLIC SERVICE AUTHORITY
DOCKET NO. 50-395
VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1
RENEWED FACILITY OPERATING LICENSE NO. NPF-12

1. The U.S. Nuclear Regulatory Commission (the Commission or the NRC) having previously made the findings set forth in License No. NPF-12 issued August 6, 1982, has now found that:

   A. The application to renew License No. NPF-12 filed by South Carolina Electric & Gas Company acting for itself and South Carolina Public Service Authority (the licensees) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. 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 Virgil C. Summer Nuclear Station (V. C. Summer), Unit 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. South Carolina Electric & Gas Company[1] is technically qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

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[1]South Carolina Electric & Gas company is authorized to act as agent for the South Carolina Public Service Authority and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

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

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the Commission concludes that the issuance of Renewed Facility Operating License NPF-12, 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; and

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

2. On the basis of the foregoing findings regarding this facility, Facility Operating License No. NPF-12, issued August 6, 1982, is superseded by Renewed Facility Operating License No. NPF-12, which is hereby issued to the South Carolina Electric & Gas Company and the South Carolina Public Service Authority (the licensees) to read as follows:

A. This renewed license applies to the Virgil C. Summer Nuclear Station, Unit 1, a pressurized water reactor and associated equipment (the facility), owned by the South Carolina Electric & Gas Company and South Carolina Public Service Authority. The facility is located in Fairfield County, South Carolina, and is described in South Carolina Electric & Gas Company's Final Safety Analysis Report, as amended through No. 33, and the Environmental Report, as amended through Amendment No. 5.

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

(1) South Carolina Electric & Gas Company (SCE&G), pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in Fairfield County, South Carolina, in accordance with the procedures and limitations set forth in this renewed license;

(2) South Carolina Public Service Authority to possess the facility at the designated location in Fairfield County, South Carolina, in accordance with the procedures and limitations set forth in this renewed license;

(3) SCE&G, 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 amounts required for reactor operation, as described in the Final Safety Analysis Report, as amended through Amendment No. 33;

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

(5) SCE&G, 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 of components; and

(6) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as m[a]y be produced by the operation of the facility.

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

(1) Maximum Power Level

SCE&G is authorized to perathe the facility at reactor core power levels not in excess of 2900 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this renewed license. The preoccupation tests, startup tests and other items identified in Attachment 1 to this renewed license shall be completed as specified. Attachment 1 is hereby incorporated into this renewed license.

(2) Technical Specifications and Environmental Protection Plant

The Technical Specifications contained in Appendix A, as revised through Amendment No. 207, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Conduct of Work Activities During Fuel Load and Initial Startup

SCE&G shall review by committee all facility construction, preoperational testing, and system demonstration activities performed concurrently with facility initial fuel loading or with the facility startup test program to assure that the activity will not affect the safe performance of the facility fuel loading or the portion of the facility startup program being performed. The review shall address, as a minimum, system interaction, span of control, staffing, security, and health physics, with respect to the performance of the activity concurrently with the facility fuel loading or the portion of the facility startup program being performed. The committee for the review shall be composed of at least three members, knowledgeable in the above areas, and who meet the qualifications for professional-technical personnel specified by Section 4.4 of ANSI N18.7-1971. At least one of these three shall be a senior member of the Plant Manager's staff.

(4) Initial Test Program

SCE&G shall conduct the post-fuel-loading initial test program set forth in Chapter 14 of the Final Safety Analysis Report, as amended through No. 33, without making any major modifications to this program unless such modifications have been identified and have received prior NRC staff approval. Major modifications are defined as:

a. Elimination of any test other than those identified as non-essential in Chapter 14 of SCE&G's Final Safety Analysis Report, as amended through Amendment No. 33,

b. Modifications of test objectives, methods or acceptance criteria for any test other than those identified as non-essential in Chapter 14 of SCE&G's Final Safety Analysis Report, as amended through Amendment No. 33,

c. Performance of any test at a power level different from the power level indicated in the described program; and

d. Failure to complete any tests included in the described program (planned or scheduled for power levels up to the authorized power level).

For the performance of startup testing as described in Table 14.1-75 of the Final Safety Analysis Report, as amended through Amendment No. 33, compliance with items 3 and 4 of Table 3.3-1 of the Technical Specifications is not required.

(5) Stability of the West Embankment and its Effects on the Intake Structure (Section 2.5.4, SSER 3)

SCE&G shall conduct the monitoring program discussed in Section 2.5.4.10.6.2 of the Final Safety Analysis Report, as amended through Amendment No. 33, to specifically include the following:

a. At the vicinity of the pumphouse and intake structure, four settlement points capable of monitoring both horizontal and vertical movements shall be established to monitor the embankment movements.

b. The submerged slope profile of the west embankment over the intake structure shall be established and monitored to detect any unusual movements that may affect the intake structure.

c. The schedule and the reporting requirements of the above inspection shall be in accordance with the recommendations stated in Regulatory Guide 1.127.

d. The condition of the intake structure shall be monitored to detect new cracks and changes to the old grouted or ungrouted cracks. Observed changes (length or width) in existing cracks and any new cracks shall be reported by SCE&G to the NRC staff. The maximum inspection interval for this monitoring of the intake structure is five (5) years.

e. The condition of the intake structure shall also be monitored as specified in (d.) above following any earthquake during which the plant seismic instrumentation indicates that the operating basis earthquake has been exceeded.

(6) Design Verification Program (Section 3.7.4, SSER #5)

Prior to December 31, 1982 SCE&G shall provide a final report to the NRC staff delineating the final resolution of the actions taken to satisfy the recommendations of the independent design verification conducted by Stone & Webster Engineering Corporation.

(7) Thermal Sleeves (Section 3.9.3, SSER #5)

Prior to startup after the first refueling outage, SCE&G shall provide, for NRC staff review and approval, justification for continued operation with the thermal sleeves removed from selected nozzles in the reactor coolant system.

(8) Environmental Qualification of Mechanical and Electrical Equipment (Section 3.11, SSER 4)

a. SCE&G shall complete all actions related to environmental qualification of equipment on a schedule specified in Section 3.11 of Supplement 4 to the Safety Evaluation Report.

b. Complete and audible records shall be available and maintained at a central location by SCE&G. Such records shall describe the environmental qualification methods used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1, dated July 1981. Such records shall be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document compliance with NUREG-0588.

c. Prior to startup after the first major shutdown or refueling outage after June 1983, SCE&G shall be in compliance with the provisions of NUREG-0588 for safety-related electrical equipment exposed to a harsh environment.

(9) Mechanical Performance (Section 4.2.3, SER)

Prior to startup after the first refueling outage, SCE&G shall examine fuel rods for baffle-jetting failure as specified in Section 4.2.3 of the Safety Evaluation Report. Should damage be observed at that time, a corrective action plan shall be submitted to the NRC staff for review and approval.

(10) Overpressurization Protection (Section 5.2.2, SSER 4)

Prior to startup after the first refueling outage, SCE&G shall install an NRC staff-approved low-temperature overpressurization protection system. The preliminary design shall be provided for NRC staff review not later than June 30, 1983.

(11) Inservice Inspection and Testing (Section 5.2.4, SSER 3)

SCE&G shall perform the following actions in conjunction with the first inservice examination:

a. Demonstrate the ability of the ultrasonic examination procedure to detect actual flaws and/or artificial reflectors in the volume subject to examination to the acceptance standards of Paragraph IWB-3500 in weldments representative of the design and materials of construction.

b. In the event that one-third thickness semi-circular reference flaws cannot be detected and discriminated from inherent anomalies, the entire volume of the weld shall be examined during the inservice inspection.

c. The reporting of the inservice inspection examination results shall be documented in a manner to define qualitatively whether, the weldment and the heat affected zone and adjacent base metal on both sides of the weld were examined by ultrasonic angle beam techniques.

(9) Design Description - Control (Section 4.3.2. SER)

SCE&G is prohibited from using part-length rods during power operation.

(13) Deleted

(14) Deleted

(15) Deleted

(16) Cable Tray Separation (Section 8.3.3, SSER 4)

Prior to startup after the first refueling outage, SCE&G shall implement the modifications to the cable trays discussed in Section 8.3.3 of Supplement No. 4 to the Safety Evaluation Report or demonstrate to the NRC staff that faults induced in non-class 1E cable trays will not result in failure of cable in the adjacent Class 1E cable trays.

(17) Alternate Shutdown System Section 9.5.1, SSER 4)

Prior to startup after the first refueling outage, SCE&G shall install a source range neutron flux monitor independent of the control complex as part of the alternate shutdown system.

(18) Fire Protection Program

South Carolina Electric & Gas Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated 11/15/11 (and supplements dated 1/26/12, 10/10/12, 2/1/13, 4/1/13, 10/14/13, 11/26/13, 1/9/14, 2/25/14, 5/2/14, 5/11/14, 8/14/14, 10/9/14, and 12/11/14) and as approved in the safety evaluation report dated 02/11/15. 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.

a. **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.

1. 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.

2. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-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.

b. **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 approve 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 approve 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 safety evaluation dated February 11, 2015. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

c. **<u>Transition License Conditions</u>**

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

2. The licensee shall implement the modifications to its facility, as described in Attachment S, Table S-1, "Plant Modifications Committed," of SCE&G letter RC-14-0196, dated December 11, 2014, by the end of the calendar year 2015. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall implement items listed in Attachment S, Table S-2, "Implementation Items," of SCE&G letter RC-14-0196, dated December 11, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by March 31, 2016 as follows:

    a. Items 3, 6, 7, 8, 10, 13, 14, 17, 19, and 21 within 180 days of NRC approval.
    b. Items 1, 2, 4, 11, and 12 by December 31, 2015.
    c. Items 5, 15, 16, 18, 20, 22, and 23 by March 31, 2016.

(19) <u>Instrument and Control Vibration Tests for Emergency Diesel Engine Auxiliary Support Systems (Section 9.5.4, SER)</u>

Prior to startup after the first refueling outage, SCE&G shall either provide test results and results of analyses to the NRC staff for review and approval which validate that the skid-mounted control panels and mounted equipment have been developed, tested, and qualified for operation under severe vibrational stresses encountered during diesel engine operation, or SCE&G shall floor mount the control panels presently furnished with the diesel generators separate from the skid on a vibration-free floor area.

(20) <u>Solid Radioactive Waste Treatment System (Section 11.2.3, SSER 4)</u>

SCE&G shall not ship "wet" solid wastes from the facility until the NRC staff has reviewed and approved the process control program for the cement solidification system.

(21) <u>Process and Effluent Radiological Monitoring and Sampling Systems (Section 11.3, SSER 4)</u>

Prior to startup after the first refueling outage, SCE&G shall install and calibrate the condensate demineralizer backwash effluent monitor RM-L11.

(22) <u>Core Reactivity Insertion Events (Section 15.2.4, SSER 4)</u>

For operations above 90% of full power, SCE&G shall control the reactor manually or the rods shall be out greater than 215 steps until written approval is received from the NRC staff authorizing removal of this restriction.

(23) <u>NUREG-0737 Conditions (Section 22)</u>

SCE&G shall complete the following conditions to the satisfaction of the NRC staff. Each item references the related subpart of Section 22 of the SER and/or its supplements.

a. <u>Procedures for Transients and Accidents (I.C.1, SSER 4)</u>

Prior to startup after the first refueling outage, SCE&G shall implement emergency operating procedures based on guidelines approved by the NRC staff.

b. Special Low Power Testing and Training (I.G.1, SSER 4)

Within twelve months following completion of the startup test program, SCE&G shall provide a report describing the results of a comparison of actual plant data taken during the natural circulation test program to the simulator responses described in the SCE&G letter, T. C. Nichols, Jr. to H. R. Denton dated March 31, 1982.

c. Direct Indication of Safety Valve Position (II.D.3, SSER 4)

Prior to exceeding 5 percent of full power, the safety valve position indication system shall be seismically qualified by SCE&G consistent with the component or system to which it is attached, and documentation of this shall be provided to the NRC staff for review and approval.

d. Inadequate Core Cooling Instruments (II.F.2, SSER 4)

Prior to startup after the first major shutdown or refueling outage after June 30, 1983, SCE&G shall complete upgrading of the incore thermocouple wiring and qualification of isolators, reference junction boxes and connectors.

e. Plant-Specific Calculations for Compliance with 10 CFR Section 50.46 (II.K.3.31, SSER 1)

Within one year after model revisions are approved by the NRC staff, SCE&G shall provide a supplemental plant-specific analysis to verify compliance with 10 CFR 50.46, using the revised models developed under item II.K.3.30 of NUREG-0737.

f. Upgrade Emergency Support Facilities (III.A.1.2, SSER 4)

SCE&G shall complete its emergency response facilities as follows:

(i) Safety parameter display system - April 1, 1983
(ii) Emergency operations facility - April 1, 1983
(iii) Technical support center - April 1, 1983

(24) Deleted

(25) Confirmatory Seismic Analysis (ASLB Partial Initial Decision, July 20, 1982, Section VI.2)

During the first year of operation, SCE&G shall successfully complete the confirmatory program on plant equipment and components within the guidelines established in the findings contained in the ASLB Partial Initial Decision dated July 20, 1982, to demonstrate to the NRC staff's satisfaction that explicit safety margins exist for each component necessary for shutdown and continued heat removal in the event of the maximum potential shallow earthquake.

(26) Plume Exposure Emergency Planning Zone (ASLB Supplemental Partial Initial Decision, August 4, 1982, Section VIII.3)

During the first year of operation, SCE&G shall assure that the plume exposure emergency planning zone has been expanded to include the Kelly Miller, Greenbriar Headstart and Chapin Elementary schools and the emergency evacuation plans have been adjusted accordingly.

(27) Transportation Planning (ASLB Supplemental Partial Initial Decision, August 4, 1982, Section VIII.4)

During the first year of operation, SCE&G shall assure that the defects in transportation planning discussed in Finding 24 of the ASLB Supplemental Partial Initial Decision dated August 4, 1982 have been remedied.

(28) Food Pathway Contamination (ASLB Supplemental Partial Initial Decision, August 4, 1982, Section VIII.6)

During the first year of operation, SCE&G shall assure that plans to implement remedial and preventive measures for consumer protection against food pathway contamination have been formulated and communicated to the agricultural community.

(29) Siren Alerting System (ASLB Supplemental Partial Initial Decision, August 4, 1982, Section VIII.7)

Prior to exceeding 5% of full power, SCE&G shall complete installation and satisfactory testing of its siren alerting system.

(30) Emergency Facilities and Staffing (ASLB Supplemental Partial Initial Decision, August 4, 1982, Section VIII.8)

SCE&G shall complete the following three items related to emergency preparedness to the satisfaction of the NRC staff, consistent with the Supplement No. 2 to the Safety Evaluation Report, page A-13:

(i) Minimum shift manning requirements
(ii) Emergency response facilities
(iii) Meteorological and dose assessment capability

(31) Final NRC Approval of Emergency Preparedness (ASLB Supplemental Partial Initial Decision, August 4, 1982, Section VIII.9)

Prior to exceeding 5% of full power, final NRC staff approval of the state of emergency preparedness for the Virgil C. Summer Nuclear Station site shall be required.

(32) Deleted

(33) Emergency Preparedness Exercise (Section 13.3, SSER #5)

Prior to March 31, 1983, SCE&G shall conduct an emergency exercise similar to that conducted on May 5, 1982 but which includes full participation of the local governments and partial State participation.

(34) Mitigation Strategy License Condition

SCE&G shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

a. Fire fighting response strategy with the following elements:
   i. Pre-defined coordinated fire response strategy and guidance
   ii. Assessment of mutual aid fire fighting assets
   iii. Designated staging areas for equipment and materials
   iv. Command and control
   v. Training of response personnel

b Operations to mitigate fuel damage considering the following:
   i. Protection and use of personnel assets
   ii. Communications
   iii. Minimizing fire spread
   iv. Procedures for implementing integrated fire response strategy
   v. Identification of readily-available pre-staged equipment
   vi. Training on integrated fire response strategy
   vii. Spent fuel pool mitigation measures

c. Actions to minimize release to include consideration of:
   i. Water spray scrubbing
   ii. Dose to onsite responders

D. An exemption to the requirements of Paragraph III.B.4 of Appendix G to 10 CFR Part 50 is described in Section 5.3.1 of Supplement No. 1 to the Office of Nuclear Reactor Regulation's Safety Evaluation Report. A limited exemption to the requirements of Section IV.F.1(b) of Appendix E to 10 CFR Part 50 is described in a letter from B. J. Youngblood, NRC to O. W. Dixon, Jr., dated November 2, 1982. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. SCE&G shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Virgil C. Summer Nuclear Station Security Plan," as updated through May 15, 2006. This document includes the Security Training and Qualification Plan as Appendix B and the Safeguards Contingency Plan as Appendix C.

SCE&G 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 SCE&G CSP was approved by License Amendment No. 198.

F. This renewed license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE&G shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

G. Reporting to the Commission:

(1) DELETED

(2) SCE&G shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

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

I. In accordance with the Commission's direction in its Statement of policy, Licensing and Regulatory Policy and Procedures for Environmental Protection; Uranium Fuel Cycle Impacts, October 29, 1982, this license is subject to the final resolution of the pending litigation involving Table S-3. See, Natural Resources Defense Council v. NRC, No. 74-1586 (April 27, 1982).

J. Additional License Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 185 are hereby incorporated into this renewed license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Additional Conditions.

K. Updated Final Safety Analysis Report

The South Carolina Electric & Gas Company Updated Final Safely Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. The

South Carolina Electric & Gas Company shall complete these activities no later than August 6, 2022, 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, 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, the South Carolina Electric & Gas Company may make changes to the programs and activities described in the supplement without prior Commission approval, provided that the South Carolina Electric & Gas Company evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

L. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC.

M. This renewed license is effective as of the date of issuance and shall expire at midnight, August 6, 2042.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Enclosures:

1. Appendix A (Technical Specifications)
2. Appendix B (Environmental Protection Plan)
3. Appendix C (Additional Conditions)

Date of Issuance: April 23, 2004

Attachment 1 to Renewed Operating License No. NPF-12

This attachment identifies certain preoperational tests, system demonstrations and other items which must be completed to the satisfaction of NRC Region II. SCE&G shall not proceed without written confirmation from NRC Region II that the following items have been completed in accordance with the conditions and schedules set forth below:

1. Prior to initial criticality, SCE&G shall complete to the satisfaction of NRC Region II the following 10 CFR Part 21 identified item:

   Correct the diesel generator slow start times attributed to fuel oil header drain down (81-29-01).

2. Prior to initial criticality, SCE&G shall complete to the satisfaction of NRC Region II the following open items:

   a. Complete the replacement of all damaged prefilters external to the containment building (82-07-03).

   b. Complete the repair and testing of diesel generator B before entering Mode 4.

3. Prior to exceeding 5% of full power, SCE&G shall complete to the satisfaction of NRC Region II the requirements of the following bulletins:

   a. Failure of gate-type valves to close against differential pressure (81-BU-02).

   b. Seismic analysis for as-built safety-related piping systems required to support operations above 5% of full power (79-BU-14).

4. Prior to exceeding 5% of full power, SCE&G shall complete to the satisfaction of NRC Region II the following 10 CFR 21 identified items:

   a. Westinghouse 3-inch gate valve closure failure (80-37-10).

   b. Westinghouse 4-inch gate valve closure failure (81-05-09).

5. Prior to exceeding 5% of full power, SCE&G shall complete to the satisfaction of NRC Region II the following open items:

   a. Correct the hydrogen recombiner high hydrogen alarm set at 6% versus proposed Technical Specification limit of 2% (80-06-07).

   b. Satisfactorily complete the capability test for Fe-55 analyses of liquid waste samples.

6. Prior to full power operation, SCE&G shall complete to the satisfaction of NRC Region II the requirements of the following bulletin:

   Audibility problems encountered during evacuation alarm in high noise area (79-BU-18).

7. Prior to full power operation, SCE&G shall complete to the satisfaction of NRC Region II the following open item:

   Resolve the problem of the reactor building temperature being greater than expected during hot functional testing (80-25-09).

NUREG-0932

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Technical Specifications

Virgil C. Summer Nuclear Station, Unit No. 1

Docket No. 50-395

Appendix "A" to
License No. NPF-12

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August 1982

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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.

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

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

AXIAL FLUX DIFFERENCE

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.4,

b. All equipment hatches are closed and sealed,

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

d. The containment leakage rates are within the limits of Specification 6.8.4.g, and

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT (COLR) is the unit specific document that provides core operating limits for the current operating reload cycle. The cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT 1-131

1.10 DOSE EQUIVALENT 1-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ICRP 30, Supplement to Part I, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

FREQUENCY NOTATION

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

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.

IDENTIFIED 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).

MASTER RELAY TEST

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 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 6.9.1.6 and 6.9.1.8.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.19 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 1.1.

PHYSICS TESTS

1.20 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

1.21 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)

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

PURGE - PURGING

1.23 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.

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2900 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 loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

SLAVE RELAY TEST

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

1.30 Not Used

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,

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.

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

VENTILATION EXHAUST TREATMENT SYSTEM

1.36 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

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

TABLE 1.1

OPERATIONAL MODES

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

TABLE 1.2

FREQUENCY NOTATION

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 3 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

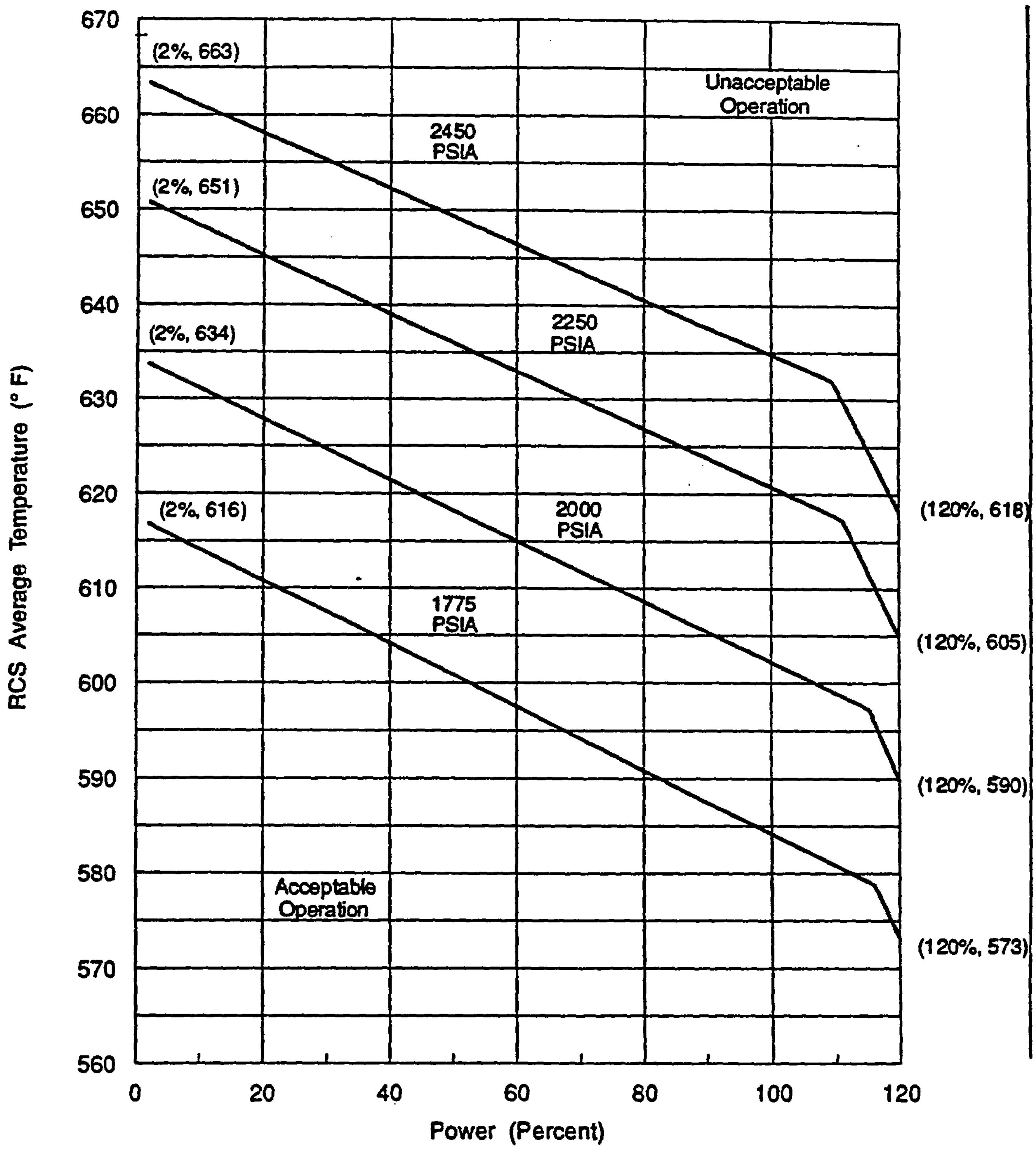
ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.



When operating in the reduced RTP region of Technical Specification 3.2.3 the restricted power level must be considered 100% RTP for this figure.

Figure 2.1-1
Reactor Core Safety Limits - Three Loop Operation

Figure 2.1-2 left blank pending NRC approval of two-loop operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation and interlocks setpoints shall be consistent with the Trip Setpoint values shown in Table 2.2-1.

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

ACTION:

a. With a reactor trip system instrumentation or interlock setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2-1 adjust the setpoint consistent with the Trip Setpoint value.

b. With the reactor trip system instrumentation or interlock setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirements of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint Value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
1.	Manual Reactor Trip	NA	NA
2.	Power Range, Neutron Flux High Setpoint Low Setpoint	 $\leq$109% of RTP $\leq$25% of RTP	 $\leq$111.2% of RTP $\leq$27.2% of RTP
3.	Power Range, Neutron Flux High Positive Rate	$\leq$5% of RTP with a time constant $\geq$2 seconds	$\leq$6.3% of RTP with a time constant $\geq$2 seconds
4.	DELETED		
5.	Intermediate Range, Neutron Flux	$\leq$25% of RTP	$\leq$31% of RTP
6.	Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7.	Overtemperature ΔT	See note 1	See note 2
8.	Overpower ΔT	See note 3	See note 4
9.	Pressurizer Pressure-Low	$\geq$1870 psig	$\geq$1859 psig
10.	Pressurizer Pressure-High	$\leq$2380 psig	$\leq$2391 psig
11.	Pressurizer Water Level-High	$\leq$92% of instrument span	$\leq$93.8% of instrument span
12.	Loss of Flow	$\geq$90% of loop design flow*	$\geq$88.9% of loop design flow*

* Loop design flow = 94,500 gpm
RTP - RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
13.	Steam Generator Water Level Low-Low Barton Transmitter Rosemount Transmitter	 $\geq$27.0% of span $\geq$27.0% of span	 $\geq$26.1% of span $\geq$25.7% of span
14.	Steam/Feedwater Flow Mis-Match Coincident With Steam Generator Water Level Low-Low Barton Transmitter Rosemount Transmitter	$\leq$40% of full steam flow at RTP $\geq$27.0% of span $\geq$27.0% of span	$\leq$42.5% of full steam flow at RTP $\geq$26.1% of span $\geq$25.7% of span
15.	Undervoltage - Reactor Coolant Pump	$\geq$4830 volts	$\geq$4760 volts
16.	Underfrequency - Reactor Coolant Pumps	$\geq$57.5 Hz	$\geq$57.1 Hz
17.	Turbine Trip A. Low Trip System Pressure B. Turbine Stop Valve Closure	 $\geq$800 psig $\geq$1% open	 $\geq$750 psig $\geq$1% open

RTP - RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
18.	Safety Injection Input from ESF	NA	NA
19.	Reactor Trip System Interlocks		
	A. Intermediate Range Neutron Flux, P-6	$\geq 7.5 \times 10^{-6}$ % indication	$\geq 4.5 \times 10^{-6}$ % indication
	B. Low Power Reactor Trips Block, P-7 a. P-10 input	$\leq$10% of RTP	$\leq$12.2% of RTP
	b. P-13 input	$\leq$10% turbine impulse pressure equivalent	$\leq$12.2% of turbine impulse pressure equivalent
	C. Power Range Neutron Flux P-8	$\leq$38% of RTP	$\leq$40.2% of RTP
	D. Low Setpoint Power Range Neutron Flux, P-10	$\geq$10% of RTP	$\geq$7.8% of RTP
	E. Turbine Impulse Chamber Pressure, P-13	$\leq$10% turbine impulse pressure equivalent	$\leq$12.2% turbine pressure equivalent
	F. Power Range Neutron Flux, P-9	$\leq$50% of RTP	$\leq$52.2% of RTP
20.	Reactor Trip Breakers	NA	NA
21.	Automatic Actuation Logic	NA	NA

RTP - RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where:	ΔT	=	Measured ΔT by RTD Instrumentation
	ΔT_o	≤	Indicated ΔT at RATED THERMAL POWER
	K_1	≤	1.23
	K_2	≥	0.0292/°F
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The funtion generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in lead-lag controller for T_{avg}, $\tau_1 \ge 28$ secs., $\tau_2 \le 4$ secs.
	T	=	Average temperature, °F
	T'	≤	Indicated T_{avg} at RATED THERMAL POWER, 572.0°F ≤ T' ≤ 587.4°F
	K_3	≥	0.00161/psi
	P	=	Pressurizer pressure, psig
	P'	≥	2235 psig, Nominal RCS operating pressure
	S	=	Laplace transform operator, sec^{-1}.

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (continued)

NOTE 1: (Continued)

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

(i) for $q_t - q_b$ between -35 percent and +6 percent $f_1(\Delta I) = 0$ where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -35 percent, the ΔT trip setpoint shall be automatically reduced by 2.46 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by 3.29 percent of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.2 percent ΔT Span.

NOTE 3: OVERPOWER ΔT

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

Where: ΔT = as defined in Note 1

ΔT_o = as defined in Note 1

K_4 ≤ 1.078

K_5 ≥ 0.02/°F for increasing average temperature and 0 for decreasing average temperature

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation

TABLE 2.2-1 (continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (continued)

NOTE 3: (continued)

τ_3	=	Time constant utilized in rate-lag controller for T_{avg}, $\tau_3 \geq 10$ secs.
K_6	$\geq$	$0.00198/°F$ for $T > T''$ and $K_6 = 0$ for $T \leq T''$
T	=	as defined in Note 1
T''	$\leq$	Indicated T_{avg} at RATED THERMAL POWER, $572.0°F \leq T'' \leq 587.4°F$
S	=	as defined in Note 1

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.3 percent ΔT Span.

PAGE 2-11
WAS DELETED PER
AMENDMENT No. 28
THAT WAS ISSUED ON
OCTOBER 12, 1984

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

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION requirements is not required.

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

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 Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODES 5 and 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

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.

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

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, as defined by Specification 4.0.2, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified 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 LCO must immediately be declared not met, and the applicable Action(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Action(s) must be entered.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing 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 and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance the applicable ASME Code and Addenda as required by 10 CFR 50, Section 50.55a.

b. Surveillance intervals specified for the inservice inspection and testing activities required by the applicable ASME Code and Addenda shall be applicable as follows in these Technical Specifications:

4.0.5 (continued)

ASME Code terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Double the Quarterly (3 month) frequency	At least once per 46 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days.

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

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

e. Nothing in the applicable ASME 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 - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% delta k/k for 3 loop operation.

APPLICABILITY: MODES 1, and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.77% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

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

*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least the following factors:

1. Reactor Coolant System boron concentration,
2. Control rod position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy production,
5. Xenon concentration, and
6. Samarium.

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.

SHUTDOWN MARGIN - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limits shown in Figure 3.1-3.

APPLICABILITY: MODES 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

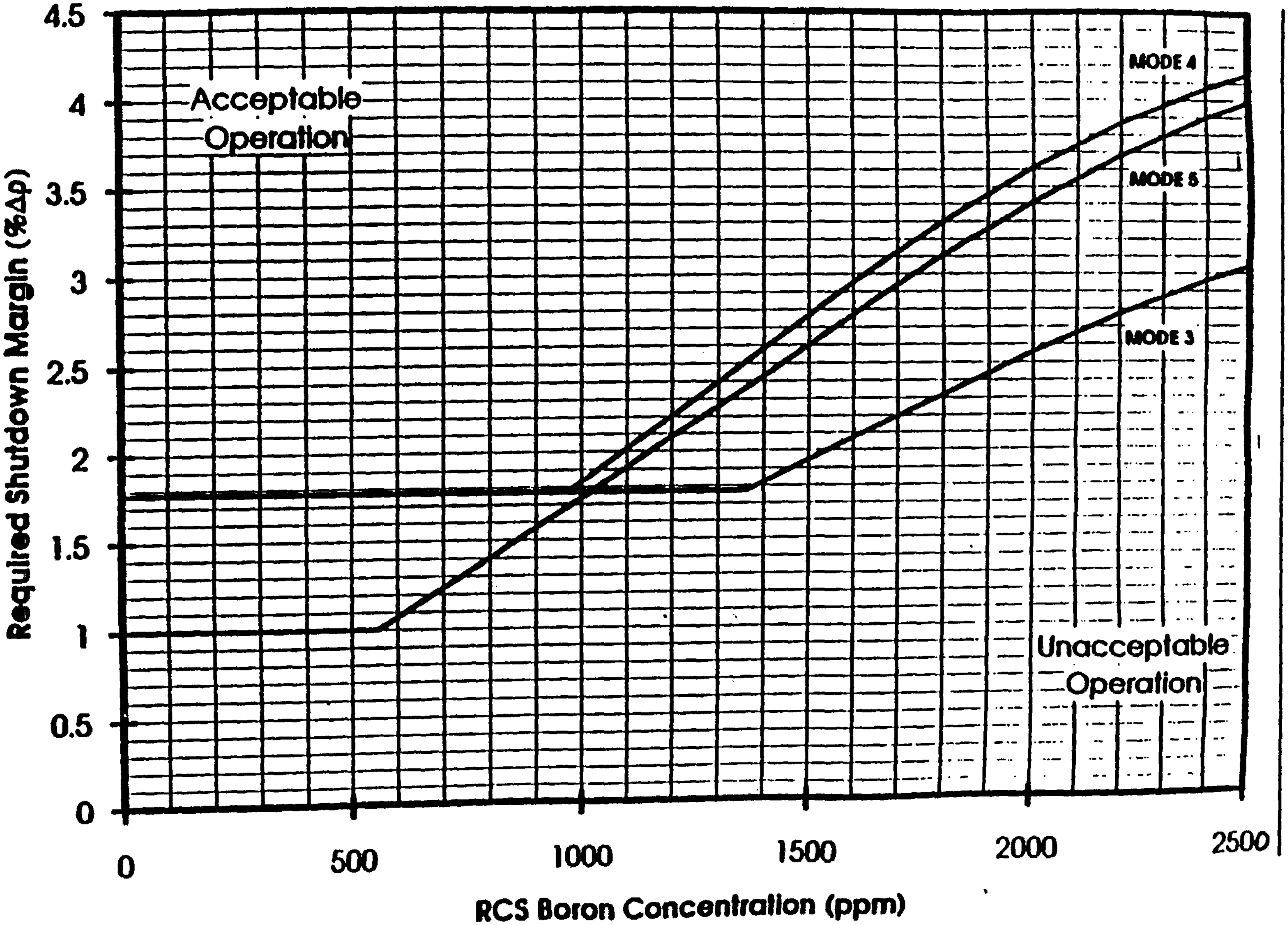
4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the required value:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. At least once per 24 hours by consideration of the following factors:

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

FIGURE 3.1-3
REQUIRED SHUTDOWN MARGIN
(MODES 3, 4, AND 5)



MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0:

APPLICABILITY: Beginning of Cycle Life (BOL) Limit - MODES 1 and 2* only#
End of Cycle Life (EOL) Limit - MODES 1, 2 and 3 only#

ACTION:

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

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

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

   3. In lieu of any other report required by Specification 6.9.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

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

*With K_{eff} greater than or equal to 1.0
#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

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

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

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

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

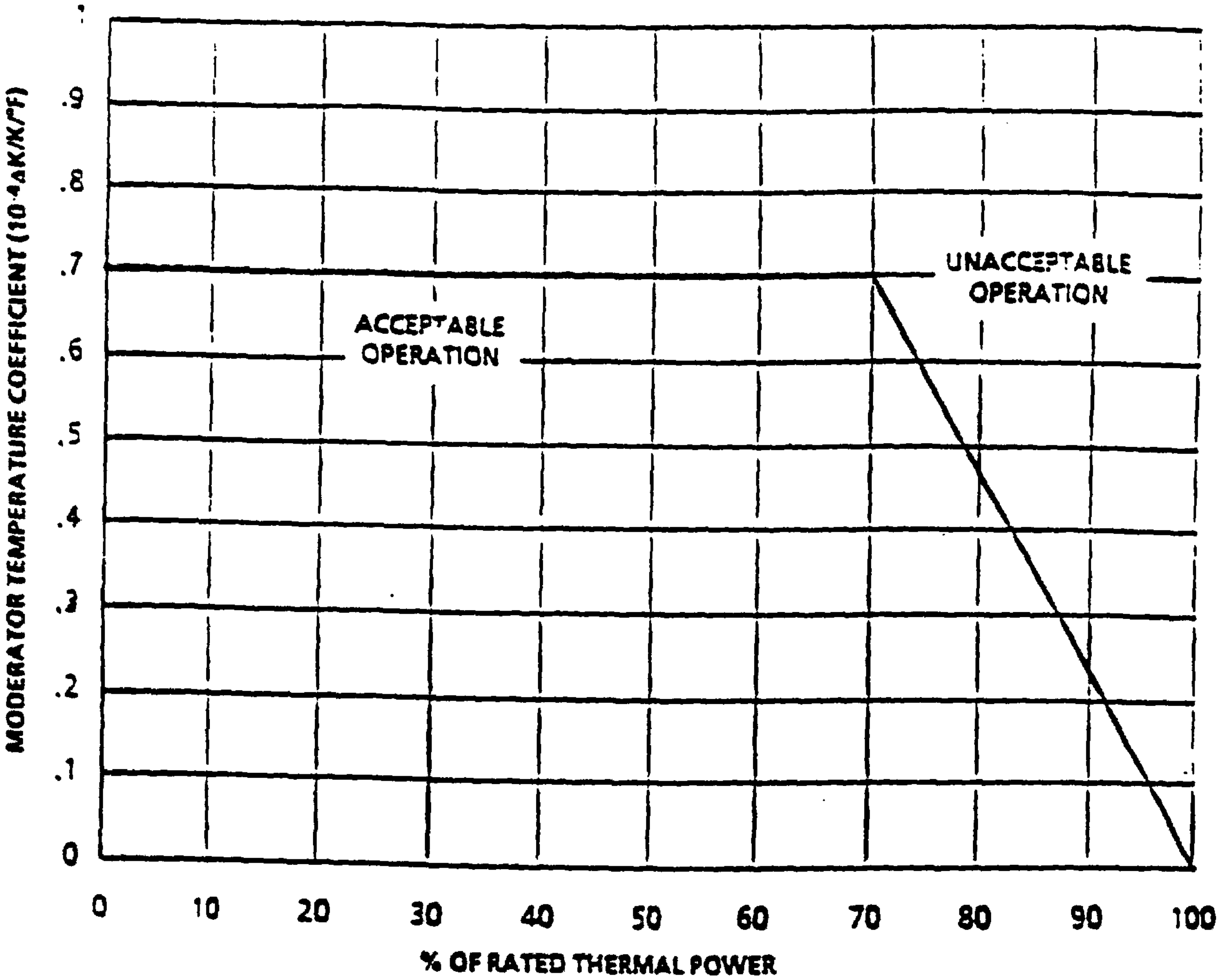


FIGURE 3.1-0
MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2#*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

#With K_{eff} greater than or equal to 1.0.
*See Special Test Exception 3.10.3.

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3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.1.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days 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.

4.1.2.1.2 Demonstrate operability of the required charging pump per Surveillance 4.5.2.f.

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. The flow path from the boric acid tanks via a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps 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 be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

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BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system with:

1. A minimum contained borated water volume of 2700 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 51,500 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

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

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage system with:

1. A minimum contained borated water volume of 14,000 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 453,800 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status 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

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

2. Verifying the contained borated water volume of the water source, and

3. Verifying the boric acid storage system solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERTION

3.1.3.1 All full length (shutdown and control) rods which are inserted in the core shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

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

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

c. With more than one full length rod inoperable due to a rod control urgent failure alarm or obvious electrical problem in the rod control system for greater than 72 hours, be in HOT STANDBY within the following 6 hours.

d. With one full length rod inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

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

    2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT (COLR); the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

    3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

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

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

c) A core power distribution measurement is obtained and $F_Q(z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

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

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control rod position indication system and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 18 months.

*With the reactor trip system breakers in the closed position.
#See Special Test Exception 3.10.5.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*#

ACTION:

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

a. Restore the rod to within the limit specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR.

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

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled Rod Group Insertion Limits versus Thermal Power For Three Loop Operation.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3
#With K_{eff} greater than or equal to 1.0.

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3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

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

a. the allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or

b. within the target band specified in the COLR about the target flux difference during base load operation.

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

ACTION:

a. For RAOC operation with the indicated AFD outside of the applicable limits specified in the COLR,

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

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AFD outside of the applicable target band about the target flux differences:

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

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the applicable RAOC limits.

*See Special Test Exception 3.10.2

**APL^{ND} is the minimum allowable power level for base load operation and will be specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.11.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE:

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

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

4.2.1.3 When in Base Load operation, the target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(z)$ shall be limited by the following relationships:

$$F_Q(z) \leq \frac{[F_Q^{RTP}]}{P} [K(z)] \text{ for } P > 0.5$$

$$F_Q(z) \leq \left[\frac{F_Q^{RTP}}{0.5}\right] [K(z)] \text{ for } P \leq 0.5$$

where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

P = $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(z) = the normalized $F_Q(z)$ for a given core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(z)$ exceeding its limit:

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

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(z)$ is demonstrated through core power distribution measurement to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the Measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and W(z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined, * or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

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

SURVEILLANCE REQUIREMENTS (Continued)

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. Increase $F_Q^M(z)$ by the appropriate penalty factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive core power distribution measurements indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationships specified in Specification 4.2.2.2.c. above not being satisfied:

1. Calculate the maximum percent over the core height (z) that $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{\left[\frac{F_Q^M(z) x W(z)}{\frac{F_Q^{RTP}}{P} x K(z)}\right] - 1\right\} x\ 100 \text{ for } P \geq 0.5$$

$$\left\{\left[\frac{F_Q^M(z) x W(z)}{\frac{F_Q^{RTP}}{0.5} x K(z)}\right] - 1\right\} x\ 100 \text{ for } P < 0.5$$

2. One of the following actions shall be taken:

   (a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the applicable AFD limits by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2.f.(1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

   (b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

   (c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

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

1. Lower core region from 0 to 10%, inclusive.
2. Upper core region from 90 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within applicable target band about the target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as the minimum value of:

$$APL^{BL} = \frac{F_Q^{RTP} \times K(z)}{F_Q^M(z) \times W(z)_{BL}} \times 100\%$$

over the core height (z) where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR. The F_Q limit is F_Q^{RTP}. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transient encountered during base load operation. F_Q^{RTP}, K(z), and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND} when the Power Distribution Monitoring System (PDMS) is inoperable; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS at any THERMAL POWER greater than APL^{ND}; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing and measurement uncertainties as specified in the COLR. The F_Q limit is F_Q^{RTP}. P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a core power distribution measurement has been obtained in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to measurement, and

2. At least once per 31 Effective Full Power Days.

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

SURVEILLANCE REQUIREMENTS (Continued)

over the core height (z) increasing since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. Increase $F_Q^M(z)$ by the appropriate penalty factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.4.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until 2 successive core power distribution measurements indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(z)$, or

2. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the maximum percent calculated over the core height (z) with the following expression:

$$\left\{\left[\frac{F_Q^M(z) x W(z)_{BL}}{\frac{F_Q^{RTP}}{P} x K(z)}\right]-1\right\} x\ 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 10%, inclusive.

2. Upper core region 90 to 100%, inclusive.

4.2.2.5 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained:

a. From a power distribution map

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. From the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

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

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled RCS Total Flow Rate Versus R For Three Loop Operation.

Where:

a. R = $\dfrac{F^N_{\Delta H}}{F^{RTP}_{\Delta H}\left[1.0 + PF_{\Delta H}(1.0 - P)\right]}$,

b. P = $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by

1. Using the movable incore detectors to obtain a power distribution map when THERMAL POWER is ≤ 25% but > 5% of RATED THERMAL POWER, or when PDMS is inoperable, and
2. Using the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER.

The measured values of $F^N_{\Delta H}$ shall be increased by the applicable $F^N_{\Delta H}$ measurement uncertainties as specified in the COLR, and used to calculate R since the RCS Total Flow Rate Versus R figure in the COLR includes measurement uncertainties of 2.1% (includes 0.1% for feedwater venturi fouling) for flow,

d. $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR, and

e. $PF_{\Delta H}$ = The Power Factor Multiplier specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation specified in the COLR:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or
2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through a core power distribution measurement and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through a core power distribution measurement and RCS total flow rate comparison, to be within the region of acceptable operation specified in the COLR prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

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

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation specified in the COLR.

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

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation specified in the COLR at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by heat balance measurement at ≥90% RATED THERMAL POWER at least once per 18 months.

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3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

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

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

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

2. Within 2 hours either:

   a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

   b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

ACTION: (Continued)

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

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

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

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

   2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

   4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

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

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

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

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

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent RATED THERMAL POWER with one Power Range Channel inoperable at least once per 12 hours by using the PDMS or movable incore detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QUADRANT POWER TILT RATIO. The incore detector monitoring shall be done with 2 sets of 4 symmetric thimbles or a full incore flux map.

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops In Operation	 2 Loops In Operation
Indicated Reactor Coolant System T_{avg}	$\leq$ 589.2°F	**
Indicated Pressurizer Pressure	$\geq$ 2206 psig*	**

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

** These values left blank pending NRC approval of two-loop operation.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by performance of the reactor trip system instrumentation surveillance requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be verified to be within its limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 2	1 1	2 2	1, 2 3*, 4*, 5*	1 9
2. Power Range, Neutron Flux					
A. High Setpoint	4	2	3	1, 2	2#
B. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##	4
B. Shutdown	2	0	1	3, 4 and 5	5
C. Shutdown	2	1	2	3*, 4*, 5*	9
7. Overtemperature ΔT					
Three Loop Operation	3	2	2	1, 2	6#
Two Loop Operation	****	****	****	****	****
8. Overpower ΔT					
Three Loop Operation	3	2	2	1, 2	6#
Two Loop Operation	****	****	****	****	****
9. Pressurizer Pressure-Low	3	2	2	1	6#
10. Pressurizer Pressure--High	3	2	2	1, 2	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1	6#
12. A. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6#
13. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	6#
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in each loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch in same loop or 2/loop-level and 1/loop-flow mismatch in same loop	1, 2	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
15. Undervoltage-Reactor Coolant Pumps	3-1/bus	2	2	1	6#
16. Underfrequency-Reactor Coolant Pumps	3-1/bus	2	2	1	6#
17. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	6#
B. Turbine Stop Valve Closure	4	4	1	1	10#
18. Safety Injection Input from ESF	2	1	2	1, 2	12

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Reactor Trip System Interlocks						
A. Intermediate Range Neutron Flux, P-6		2	1	2	2##	7
B. Low Power Reactor Trips Block, P-7	P-10 Input	4	2	3	1	7
	P-13 Input	2	1	2	1	7
C. Power Range Neutron Flux, P-8		4	2	3	1	7
D. Power Range Neutron Flux, P-10		4	2	3	1, 2	7
E. Turbine First Stage Pressure, P-13		2	1	2	1	7
F. Power Range Neutron Flux, P-9		4	2	3	1	7
20. Reactor Trip Breakers		2	1	2	1, 2	8, 11
		2	1	2	3*, 4*, 5*	9
21. Automatic Trip Logic		2	1	2	1, 2	12
		2	1	2	3*, 4*, 5*	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

The provisions of Specification 3.0.4 are not applicable.

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

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

**** Values left blank pending NRC approval of 2 loop operation.

ACTION STATEMENTS

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

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

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

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

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, 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 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE, one channel may be bypassed for up to 2 hours for maintenance on the undervoltage or shunt trip mechanisms, provided the other channel is OPERABLE, and one channel may be bypassed for up to 4 hours for concurrent surveillance testing of the Reactor Trip Breaker and automatic trip logic, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

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

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 12 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

	FUNCTIONAL UNIT	RESPONSE TIME
1.	Manual Reactor Trip	Not Applicable
2.	Power Range, Neutron Flux	$\leq$ 0.5 seconds[1]
3.	Power Range, Neutron Flux, High Positive Rate	Not Applicable
4.	Deleted	
5.	Intermediate Range, Neutron Flux	Not Applicable
6.	Source Range, Neutron Flux	$\leq$ 0.5 seconds[1]
7.	Overtemperature ΔT	$\leq$ 8.5 seconds[1][2]
8.	Overpower ΔT	$\leq$ 8.5 seconds[1][2]
9.	Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10.	Pressurizer Pressure--High	$\leq$ 2.0 seconds
11.	Pressurizer Water Level--High	Not Applicable

(1) Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

(2) The 8.5 second response time includes a 5.0 second delay for the RTDs mounted in thermowells.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. A. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 seconds
13. Steam Generator Water Level--Low-Low	$\leq$ 2.0 seconds
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Not Applicable
15. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.5 seconds
16. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 seconds
17. Turbine Trip	
A. Low Fluid Oil Pressure	Not Applicable
B. Turbine Stop Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Trip System Interlocks	Not Applicable
20. Reactor Trip Breakers	Not Applicable
21. Automatic Trip Logic	Not Applicable

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Deleted						
5. Intermediate Range, Neutron Flux	S	R(4)	S/U(1),	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4)	S/U(1),Q(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1
12. Loss of Flow	S	R	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	Q	N.A.	N.A.	1, 2
14. Steam Generator Water Level - Low Coincident with Steam/Feedwater Flow Mismatch	S	R	Q	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
17. Turbine Trip						
A. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
B. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A	1, 2
19. Reactor Trip System Interlocks						
A. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2##
B. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
C. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1

Correction Letter of 8-6-91

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
D. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
E. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
F. Low Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 12)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(13), R(14)	N.A.	1, 2, 3*, 4*, 5*

Correction letter of 8-6-91

TABLE 4.3-1 (Continued)

TABLE NOTATION

*	-	With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.
##	-	Below P-6 (Intermediate Range Neutron Flux Interlock) setpoint.
###	-	Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.
(1)	-	If not performed in previous 31 days.
(2)	-	Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(3)		Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(4)	-	Neutron detectors may be excluded from CHANNEL CALIBRATION.
(5)	-	Detector plateau curves shall be obtained evaluated and compared to manufacturer's data. For the Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(6)	-	Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(7)	-	Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(8)	-	DELETED
(9)	-	Quarterly Surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.
(10)	-	Setpoint verification is not required.
(11)	-	The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).
(12)	-	The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.
(13)	-	Local manual shunt trip prior to placing breaker in service.
(14)	-	Automatic undervoltage trip.

3/4 3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value Column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to its OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

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

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the engineered safety feature actuation system instrumentation surveillance requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

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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, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Reactor Building Pressure - High-1	3	2	2	1, 2, 3	24*
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	24*
e. Differential Pressure Between Steam Lines - High	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line	1, 2, 3	24*

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
f. Steam Line Pressure-Low	1 pressure/ loop	1 pressure and 2 loops	1 pressure and 2 loops	1, 2, 3##	24*
2. REACTOR BUILDING SPRAY					
a. Manual	2 sets - 2 switches/set	1 set	2 sets	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Reactor Building Pressure--High-3 (Phase 'A' isolation aligns spray system discharge valves and NaOH tank suction valves)	4	2	3	1, 2, 3	16

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
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	18
2) Safety Injection	See 1 above for all safety injection initiating functions and requirements.				
3) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Phase "B" Isolation					
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
2) Reactor Building Pressure--High-3	4	2	3	1, 2, 3	16
c. Purge and Exhaust Isolation					
1) Safety Injection	See 1 above for all safety injection initiating functions and requirements.				
2) Containment Radio-activity- High	2*	1	2*	1, 2, 3, 4	17
3) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17

*Purge exhaust monitor not required when purge exhaust is closed.

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
4. STEAM LINE ISOLATION					
a. Manual					
i. One Switch/line	1/steam line	1/steam line	1/operating steam line	1, 2, $3^{\#\#\#}$	23
ii. One Switch/all lines	1	1	1	1, 2, $3^{\#\#\#}$	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, $3^{\#\#\#}$	21
c. Reactor Building Pressure--High-2	3	2	2	1, 2, $3^{\#\#\#}$	24*
d. Steam Flow in Two Steam Lines--High	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, $3^{\#\#\#}$	24*
COINCIDENT WITH T_{avg}--Low-Low	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops	1, 2, $3^{\#\#\#}$	24*

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
e. Steam Line Pressure-Low	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 2 loops	1, 2, 3##,###	24*
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level-- High-High	3/loop	2/loop in any operating loop	2/loop in each oper- ating loop	1, 2	24*
b. Automatic Actuation Logic and Actuation Relay	2	1	2	1, 2	25

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
6. EMERGENCY FEEDWATER					
a. Manual Initiation	1 per pump	1 per pump	1 per pump	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Stm. Gen. Water Level-Low-Low					
i. Start Motor-Driven Pumps	3/stm. gen.	2/stm. gen. any stm gen.	2/stm. gen.	1, 2, 3	24*
ii. Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen	1, 2, 3	24*
d. Undervoltage-both ESF Busses Start Turbine-Driven Pump	2-1/bus	2	2	1, 2, 3	19
e. S.I. Start Motor-Driven Pumps	See 1 above (all S.I. initiating functions and requirements)				
f. Undervoltage-one ESF bus Start Motor-Driven Pumps	2-1/bus	1	2	1, 2	22
g. Trip of Main Feedwater Pumps Start Motor-Driven Pumps	3-1/pump	3-1/pump	3-1/pump	1, 2	19
h. Suction Transfer on Low Pressure	4	2	3	1, 2, 3	16

Correction letter of 8-6-91

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. LOSS OF POWER					
a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	2-1/bus	1	2	1, 2, 3, 4	18
b. 7.2 kv Emergency Bus Undervoltage (Degraded Voltage)	2-1/bus	1	2	1, 2, 3, 4	18
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP					
a. RWST level low-low	4	2	3	1, 2, 3	16
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	3	2	2	1, 2, 3###	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) setpoint.

Except when below P-12 with all MSIVs and bypasses closed and disabled.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - DELETED

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the inoperable channel is placed in bypass and the Minimum Channels OPERABLE requirement is met. Restore the inoperable channel to OPERABLE status within 72 hours otherwise;

Be in at least HOT STANDBY within the next 6 hours <u>and</u> in COLD SHUTDOWN within the following 30 hours.

One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

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

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

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

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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

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

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

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

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
1.	SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER.		
	a. Manual Initiation	NA	NA
	b. Automatic Actuation Logic	NA	NA
	c. Reactor Building Pressure-High 1	≤3.6 psig	≤3.86 psig
	d. Pressurizer Pressure--Low	≥1850 psig	≥1839 psig
	e. Differential Pressure Between Steamlines--High	≤97 psig	≤106 psi
	f. Steamline Pressure--Low	≥675 psig	≥635 psig(1)
2.	REACTOR BUILDING SPRAY		
	a. Manual Initiation	NA	NA
	b. Automatic Actuation Logic and Actuation Relays	NA	NA
	c. Reactor Building Pressure-High 3 (Phase 'A' isolation aligns spray system discharge valves and NaOH tank suction valves)	≤12.05 psig	≤12.31 psig

(1) Time constants utilized in lead lag controller for steamline pressure-low are as follows:
$\tau_1 \geq$ 50 secs. $\tau_2 \leq$ 5 secs.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
3.	CONTAINMENT ISOLATION		
	a. Phase "A" Isolation		
	1. Manual	NA	NA
	2. Safety Injection	See 1 above for all safety injection setpoints	See 1 above for all allowable values
	3. Automatic Actuation Logic and Actuation Relays	NA	NA
	b. Phase "B" Isolation		
	1. Automatic Actuation Logic and Actuation Relays	NA	NA
	2. Reactor Building Pressure-High 3	$\leq$12.05 psig	$\leq$12.31 psig
	c. Purge and Exhaust Isolation		
	1. Safety Injection	See 1 above for all safety injection setpoints	See 1 above for all allowable values
	2. Containment Radioactivity High	*	*
	3. Automatic Actuation Logic and Actuation Relays	NA	NA

* Trip setpoints shall be set to ensure that the limits of ODCM Specification 1.2.2.1 are not exceeded.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
4.	STEAM LINE ISOLATION		
	a. Manual	NA	NA
	b. Automatic Actuation Logic and Actuation Relays	NA	NA
	c. Reactor Building Pressure-High 2	$\leq$6.35	$\leq$6.61
	d. Steam Flow in Two Steamlines-High, Concident with	$\leq$ a function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load	$\leq$ a function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 114.0% of full steam flow at full load
	T_{avg} - Low-Low	$\geq$552.0°F	$\geq$548.4°F
	e. Steamline Pressure-Low	$\geq$675 psig	$\geq$635 psig(1)

(1) Time constants utilized in lead lag controller for steamline pressure low are as follows:
$\tau_1 \geq 50$ secs. $\tau_2 \leq 5$ secs.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
5.	TURBINE TRIP AND FEEDWATER ISOLATION		
	a. Steam Generator Water Level - High-High Barton Transmitter Rosemount Transmitter	 ≤79.2% of span ≤79.2% of span	 ≤81.0% of span ≤81.0% of span
	b. Automatic Actuation Logic	NA	NA
6.	EMERGENCY FEEDWATER		
	a. Manual	NA	NA
	b. Automatic Actuation Logic	NA	NA
	c. Steam Generator Water Level - Low-Low Barton Transmitter Rosemount Transmitter	 ≥27.0% of span ≥27.0% of span	 ≥26.1% of span ≥25.7% of span
	d. & f. Undervoltage-ESF Bus	≥ 5760 Volts with a ≤0.25 second time delay ≥ 6576 Volts with a ≤3.0 second time delay	≥ 5652 Volts with a ≤0.275 second time delay ≥ 6511 Volts with a ≤3.3 second time delay

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
	e. Safety Injection	See 1 above (all SI Setpoints)	See 1 above (all SI Setpoints)
	g. Trips of Main Feedwater Pumps	NA	NA
	h. Suction transfer on Low Pressure	$\geq$442 ft. 4 in. (2)	$\geq$441 ft. 3 in.
7.	LOSS OF POWER		
	a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	$\geq$5760 volts with a $\leq$0.25 second time delay	$\geq$5652 volts with a $\leq$0.275 second time delay
	b. 7.2 kv Emergency Bus Undervoltage	$\geq$6576 volts with a $\leq$3.0 second time delay	$\geq$6511 volts with a $\leq$3.3 second time delay
8.	AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP		
	a. RWST Level Low-Low	$\geq$18%	$\geq$15%
	b. Automatic Actuation Logic and Actuation Relays	NA	NA

(2) Pump suction head at which transfer is initiated is stated in effective water elevation in the condensate storage tank.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
9.	ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
	INTERLOCKS		
	a. Pressurizer Pressure, P-11	1985 psig	≥1974 psig & ≤1996 psig
	b. T_{avg} Low-Low, P-12	552°F	≥548.4°F & ≤555.6°F
	c. Reactor Trip, P-4	NA	NA

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection	Not Applicable
b. Reactor Building Spray	Not Applicable
c. Containment Isolation Phase "A" Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
e. Feedwater Isolation	Not Applicable
f. Emergency Feedwater	Not Applicable
g. Essential Service Water	Not Applicable
h. Reactor Building Cooling Fans	Not Applicable
i. Control Room Isolation	Not Applicable
2. Reactor Building Pressure-High	
a. Safety Injection (ECCS)	$\leq 27.0^{(2)}/27.0^{(1)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	$\leq 45.0^{(4)}/55.0^{(5)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
	e.	Reactor Building Purge and Exhaust Isolation	Not Applicable
	f.	Emergency Feedwater Pumps	Not Applicable
	g.	Service Water System	≤71.5(4)/81.5(5)
	h.	Reactor Building Cooling Units	≤76.5(4)/86.5(5)
	i.	Control Room Isolation	Not Applicable
3.	Pressurizer Pressure-Low		
	a.	Safety Injection (ECCS)	≤27.0(2)/27.0(1)
	b.	Reactor Trip (from SI)	≤3.0
	c.	Feedwater Isolation	≤10.0
	d.	Containment Isolation -Phase "A"	≤45.0(4)/55.0(5)
	e.	Reactor Building Purge and Exhaust Isolation	Not Applicable
	f.	Emergency Feedwater Pumps	Not Applicable
	g.	Service Water System	≤71.5(4)/81.5(5)
	h.	Reactor Building Cooling Units	≤76.5(4)/86.5(5)
	i.	Control Room Isolation	Not Applicable
4.	Differential Pressure Between Steam Lines-High		
	a.	Safety Injection (ECCS)	≤27.0(2)/37.0(3)
	b.	Reactor Trip (from SI)	≤3.0
	c.	Feedwater Isolation	≤10.0
	d.	Containment Isolation -Phase "A"	≤45.0(4)/55.0(5)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
	e.	Reactor Building Purge and Exhaust Isolation	Not Applicable
	f.	Emergency Feedwater Pumps	Not Applicable
	g.	Service Water System	$\leq 71.5^{(4)}/81.5^{(5)}$
	h.	Reactor Building Cooling Units	$\leq 76.5^{(4)}/86.5^{(5)}$
	i.	Control Room Isolation	Not Applicable
5.	Steam Line Pressure-Low		
	a.	Safety Injection - ECCS	$\leq 27.0^{(2)}/37.0^{(3)}$
	b.	Reactor Trip (from SI)	≤ 3.0
	c.	Feedwater Isolation	≤ 10.0
	d.	Containment Isolation - Phase "A"	$\leq 45.0^{(4)}/55.0^{(5)}$
	e.	Reactor Building and Purge and Exhaust Isolation	Not Applicable
	f.	Emergency Feedwater Pumps	Not Applicable
	g.	Service Water System	$\leq 71.5^{(4)}/81.5^{(5)}$
	h.	Reactor Building Cooling Units	$\leq 76.5^{(4)}/86.5^{(5)}$
	i.	Steam Line Isolation	≤ 10.0
	j.	Control Room Isolation	Not Applicable
6.	Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low		
	a.	Steam Line Isolation	≤ 12.0
7.	Reactor Building Pressure-High-2		
	a.	Steam Line Isolation	≤ 9.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Reactor Building Pressure--High-3	
a. Reactor Building Spray	$\leq$ 42.0(4)/52.0(5)
b. Containment Isolation-Phase "B"	Not Applicable
9. Steam Generator Water Level--High-High	
a. Turbine Trip	Not Applicable
b. Feedwater Isolation	$\leq$ 13.0
10. Steam Generator Water Level - Low-Low	
a. Motor-driven Emergency Feedwater Pumps	$\leq$ 60.0
b. Turbine-driven Emergency Feedwater Pumps	$\leq$ 60.0
11. Undervoltage - Both ESF Busses	
a. Turbine-driven Emergency Feedwater Pumps	$\leq$ 60.0
12. Undervoltage-one ESF Bus	
a. Motor-driven Emergency Feedwater Pumps	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
13.	Trip of Main Feedwater Pumps	
	a. Motor-driven Emergency Feedwater Pumps	Not Applicable
14.	Loss of Power	
	a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	≤10.3
	b. 7.2 kv Emergency Bus Undervoltage (Degraded Voltage)	≤13.3
15.	Containment Radioactivity--High	
	a. Purge and Exhaust Isolation	Not Applicable
16.	RWST Level--Low-Low	
	a. Automatic Switchover to Containment Sump	Not Applicable
17.	Aux Feed Suction Pressure Low	
	a. Suction transfer	Not Applicable
Note:	Response time for Motor-driven Emergency Feedwater Pumps on all SI signal starts	≤60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays from under voltage included. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps and RHR pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included.

(2) Diesel generator starting delay not included. Sequence loading delay included. Offsite power available. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(3) Diesel generator starting and sequence loading delays from under voltage included. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then the VCT valves close) is included.

(4) Diesel generator starting delay not included. Sequence loading delay included. Offsite power available.

(5) Diesel generator starting and sequence loading delays from undervoltage included.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2, 3, 4
c. Reactor Building Pressure-High-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. REACTOR BUILDING SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2, 3, 4
c. Reactor Building Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continue)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. CONTAINMENT ISOLATION								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
3) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2, 3, 4
b. Phase "B" Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2, 3, 4
2) Reactor Building Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2, 3, 4
2) Containment Radioactivity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
3) Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. STEAM LINE ISOLATION								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2, 3
c. Reactor Building Pressure-High-2	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
with T_{avg}--Low-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION								
a. Steam Generator Water Level--High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2
6. EMERGENCY FEEDWATER								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
EMERGENCY FEEDWATER (Continued)								
d. Undervoltage - Both ESF Busses	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
f. Undervoltage - One ESF Bus	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
h. Suction transfer on low pressure	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. LOSS OF POWER								
a. 7.2 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 7.2 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP								
a. RWST level low-low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R(3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS								
a. Pressurizer Pressure, P-11	N.A.	R.	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low, Low T_{avg}, P-12	N.A.	R.	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) The 36 inch containment purge supply and exhaust isolation valves are sealed closed during Modes 1 through 4, as required by TS 3.6.1.7. With these valves sealed closed, their ability to open is defeated; therefore, they are excluded from the quarterly slave relay test.

(3) Slave Relay Testing will be conducted every 18 months for Westinghouse type AR relays and preferably during a refueling outage to preclude the risk of actuation. Replacement relays other than Westinghouse type AR or reconciled Cutler-Hammer relays will require further analysis and NRC approval to maintain the established frequency.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

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 Specifications 3.0.3 and 3.0.4 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 ANALOG CHANNEL OPERATIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^4 mR/hr	25
b. Deleted					
2. PROCESS MONITORS					
a. Deleted					
b. Containment					
i. Deleted					
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^6 cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^6 cpm	29

* With fuel in the storage pool or building

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
PROCESS MONITORS (Continued)					
d. Noble Gas Effluent Monitors (High Range)					
i. Main Plant Vent (RM-A13)	1	1, 2, 3 & 4	N/A	0.1 - 10^7 mR/hr	30
ii. Main Steam Line (RM-G19A, B, C	1/steam line	1, 2, 3 & 4	N/A	0.1 - 10^7 mR/hr	30
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	1	1, 2, 3 & 4	N/A	0.1 - 10^7 mR/hr	30

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 25 - 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 26 - 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 27 - Deleted

ACTION 28 - Deleted

ACTION 29 - 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 emergency mode of operation.

ACTION 30 - With the number of OPERABLE channels 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.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. Deleted				
2. PROCESS MONITORS				
a. Deleted				
b. Containment				
i. Deleted				
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	ALL MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

* With fuel in the storage pool or building

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO using a full-core flux map per Specification 4.2.4.2, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE at least once per 24 hours, by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

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METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

CHANNEL	INSTRUMENT DESIGNATION & LOCATION	MINIMUM OPERABLE
1. Wind Speed		
	a. Wind Speed Lower - Primary Met Tower 10m	
	b. Wind Speed Upper - Primary Met Tower 61m	2
2. Wind Direction		
	a. Wind Direction Lower Primary Met Tower 10m	
	b, Wind Direction Upper Primary Met Tower 61m	2
3. Atmospheric Stability		
	a. Delta T 1 Primary Met Tower -10-61m	
	b. Delta T 2 Primary Met Tower -10-40m	2

Elevations nominal above grade elevation

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Wind Speed Lower 10m	D	SA
b. Wind Speed Upper 61m	D	SA
2. Wind Direction		
a. Wind Direction Lower 10m	D	SA
b. Wind Direction Upper 61m	D	SA
3. Atmospheric Stability		
a. Delta T 1 10-61m	D	SA
b. Delta T 2 10-40m	D	SA

Elevations nominal above grade elevation

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:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	Reactor Trip Switchgear	OPEN-CLOSE	1/trip breaker
2. Pressurizer Pressure	CREP	0-3000 psig	1
3. Pressurizer Level	CREP	0 - 100%	1
4. Steam Generator Pressure	CREP	0 - 1300 psig	1/steam generator
5. Steam Generator Level	CREP	0 - 100%	1/steam generator
6. Condensate Storage Tank Level	CREP	0 - 40 feet	1
7. Reactor Coolant System Hot Leg Temperature	CREP	0 - 700°F	1/loop
8. Reactor Coolant System Cold Leg Temperature	CREP	0 - 700°F	1/loop
9. Reactor Coolant System Pressure	CREP	0-3000 psig	1
10. Pressurizer Relief Tank Level	CREP	0-100%	1
11. Reactor Building Temperature	CREP	50°-350°F	1
12. Boric Acid Tank Level	CREP	0-100%	1/boric acid tank

CREP - Control Room Evacuation Panel

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N. A.
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Pressure	M	R
5. Steam Generator Level	M	R
6. Condensate Storage Tank Level	M	R
7. Reactor Coolant System Hot Leg Temperature	M	R
8. Reactor Coolant System Cold Leg Temperature	M	R
9. Reactor Coolant System Pressure	M	R
10. Pressurizer Relief Tank Level	M	R
11. Reactor Building Temperature	M	R
12. Boric Acid Tank Level	M	R

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY MODES 1, 2, and 3.

ACTION:

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

b.1 With the number of OPERABLE Reactor Building radiation monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

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

ii) 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.

b.2 Deleted

b.3 With the number of OPERABLE accident monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a monthly CHANNEL CHECK and a CHANNEL CALIBRATION every refueling outage. The Reactor Building Radiation Level Instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for the range decades above 10R/hr and a single point calibration of the detector below 10R/hr with an installed or portable gamma source.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Reactor Building Pressure - Narrow Range Instrument Loop/Indicator: Channel D IPT-951/IPI-951 Channel B IPT-952/IPI-952	2	1
2.	Reactor Building Pressure - Wide Range Instrument Loop/Indicator: Channel D IPT-954A/IPI-954A Channel E IPT-954B/IPI-954B	2	1
3.	Reactor Building Radiation Level - High Range Instrument Loop/Indicator: Channel A RMG-18 Channel B RMG-7	2	1
4.	Deleted		
5.	Reactor Building/RHR Sump Level Instrument Loop/Indicator: Channel A ILT-1969/ILI-1969 Channel B ILT-1970/ILI-1970	2	1
6.	Reactor Coolant Outlet Temperature - T_{Hot} - Wide Range Instrument Loop/Indicator: Channel A ITE-413/ITI-413 Channel A ITE-423/ITI-423 Channel E ITE-433/ITR-413	2	1
7.	Reactor Coolant Inlet Temperature - T_{Cold} - Wide Range Instrument Loop/Indicator: Channel E ITE-410/ITI-410 Channel E ITE-420/ITI-420 Channel E ITE-430/ITR-410	2	1

TABLE 3.3-10 (continued)

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
8.	Reactor Coolant Pressure - Wide Range Instrument Loop/Indicator: Channel E IPT-402/IPI-402 Channel A IPT-403/IPI-403	2	1
9.	Pressurizer Water Level Instrument Loop/Indicator: Channel A ILT-459/ILI-459A Channel D ILT-460/ILI-460 Channel B ILT-461/ILI-461	2	1
10.	Reactor Coolant System Subcooling Margin Instrument Loop/Indicator: Channel A ITM-499A Channel B ITM-499B	2	1
11.	Reactor Vessel Level Instrument Loop/Indicator: Channel A ILT-1311/ILI-1311, ILT-1312/ILI-1312 Channel B ILT-1321/ILI-1321, ILT-1322/ILI-1322	2	1
12.	Core Exit Temperature Instrument Loop/Indicator: Channel A ITEs 2, 4, 9, 12, 13, 15, 19, 21, 22, 23, 24, 25, 26, 27, 28, 29, 31, 32, 33, 35, 39, 41, 42, 45, 46, 47 Channel B ITEs 1, 3, 5, 6, 7, 8, 10, 11, 14, 16, 17, 18, 20, 30, 34, 36, 37, 38, 40, 43, 44, 48, 49, 50, 51 (Primary display is the plant computer) (Backup displays are ITM 499 A&B)	4/core quadrant/ channel	2/core quadrant/ channel
13.	Neutron Flux Instrument Loop/Indicator: Channel 1 INI-35 Channel 2 INI-36	2	1

TABLE 3.3-10 (continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
14. Steam Line Pressure Instrument Loop/Indicator: SG A IPTs-474, 475, 476/IPIs-474, 475, 476 SG B IPTs-484, 485, 486/IPIs-484, 485, 486 SG C IPTs-494, 495, 496/IPIs-494, 495, 496	2/stm. gen.	1/stm. gen.
15. Steam Generator Water Level - Wide Range Instrument Loop/Indicator: SG A ILT-477/ILI-477 SG B ILT-487/ILI-487 SG C ILT-497/ILI-497	1/stm. gen.	1/stm. gen.
16. Steam Generator Water Level - Narrow Range Instrument Loop/Indicator: SG A ILTs 474, 475, 476/ILIs 474, 475, 476 SG B ILTs 484, 485, 486/ILIs 484, 485, 486 SG C ILTs 494, 495, 496/ILIs 494, 495, 496	2/stm. gen.	1/stm. gen.
17. Emergency Feedwater Flow Instrument Loop/Indicator: Channel A SG A IFT-3561/IFI-3561 SG B IFT-3571/IFI-3571 SG C IFT-3581/IFI-3581 Channel B SG A IFT-3561A/IFI-3561B SG B IFT-3571A/IFI-3571B SG C IFT-3581A/IFI-3581B	2/stm. gen.	1/stm. gen.
18. Refueling Water Storage Tank Level Instrument Loop/Indicator: Channel A ILT-990/ILI-990 Channel B ILT-992/ILI-992	2	1

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INSTRUMENTATION

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The explosive gas monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With an explosive gas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, declare the channel inoperable and take the ACTION shown in Table 3.3-13.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to operable status within 30 days and, if unsuccessful prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 to explain why this inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Oxygen Monitor	2	**	44
b. Hydrogen Monitor	1	**	42

TABLE 3.3-13 (Continued)

TABLE NOTATION

**During waste gas holdup system operation (treatment for primary system offgases).

ACTION 38 - (Not used)

ACTION 39 - (Not used)

ACTION 40 - (Not used)

ACTION 41 - (Not used)

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

ACTION 43 - (Not used)

ACTION 44 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both the channels inoperable, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
a. Hydrogen Monitor	D	Q(1)	M	**
b. Oxygen Monitor	D	Q(2)	M	**

TABLE 4.3-9 (Continued)

TABLE NOTATION

** During waste gas holdup system operation (treatment for primary system offgases).

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

1. 1500 ± 30 ppm hydrogen, balance nitrogen, for the outlet hydrogen monitor and
2. 4 ± 0.1 volume percent hydrogen, balance nitrogen, for the inlet hydrogen monitor.

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

1. 75 ± 1.5 ppm oxygen, balance nitrogen, for the outlet oxygen monitor and
2. 3.5 ± 0.1 volume percent oxygen, balance nitrogen, for the inlet oxygen monitor.

INSTRUMENTATION

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With one or more loose part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

Correction letter of 12-9-91

LIMITING CONDITION FOR OPERATION

3.3.3.11 The Power Distribution Monitoring System (PDMS) shall be OPERABLE with:

a. A minimum of the following inputs from the plant available for use by the PDMS as defined in Table 3.3-14.

1. Control Bank Position
2. T_{cold}
3. Reactor Power Level
4. NIS Power Range Detector Section Signals

b. Core Exit Thermocouples (T/C) meeting the criteria:

1. At least 25% operable T/C with at least 2 T/C per quadrant, and
2. The T/C pattern has coverage of all interior fuel assemblies (no face along the baffle), within a chess knight's move, radially, from a responding, calibrated T/C, or
3. At least 25% operable T/C with at least 2 T/C per quadrant, and the installed PDMS calibration was determined within the last 31 Effective Full Power Days (EFPD).
4. The T/C temperatures used by the PDMS are calibrated via cross calibration with the loop temperature measurement RTDs, and using the T/C flow mixing factors determined during installed PDMS calibration.

c. An installed PDMS calibration satisfying the criteria:

1. The initial calibration in each operating cycle is determined using measurements from at least 75% of the incore movable detector thimbles obtained at a THERMAL POWER greater than 25% of RATED THERMAL POWER.
2. The calibration is determined using measurements from at least 50% of the incore movable detector thimbles at any time except as specified in 3.3.3.11.c.1, and
3. The calibration is determined using a minimum of 2 detector thimbles per core quadrant.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, or 3.3.3.11.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2, to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11.1 The operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, and 3.3.3.11.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.11.2 Calibration of the PDMS is required:

a. at least once every 180 Effective Full Power Days when the minimum number and core coverage criteria as defined in 3.3.3.11.b.1 and 3.3.3.11.b.2 are satisfied, or

b. at least once every 31 Effective Full Power Days when only the minimum number criterion as defined in 3.3.3.11.b.3 is satisfied.

TABLE 3.3-14

REQUIRED PDMS PLANT INPUT INFORMATION

	PLANT INPUT INFORMATION	AVAILABLE INPUTS	MINIMUM NO. OF VALID INPUTS	APPLICABLE MODES
1.	Control Bank Position	4	4[a]	1[c]
2.	T_{cold}	3	2	1[c]
3.	Reactor Power Level	3	1[b]	1[c]
4.	NIS Power Range Excore Detector Section Signals	8	6[d]	1[c]

TABLE NOTATIONS

a. Determined from either valid Demand Position or the average of the valid individual RCCA position indications for all RCCAs in the Control Bank.

b. Determined from either the reactor THERMAL POWER derived using a valid secondary calorimetric measurement, the average NIS Power Range Detector Power, or the average RCS Loop ΔT.

c. Greater than 25% RTP.

d. Comprised of an upper and lower detector section signal per Power Range Channel; a minimum of 3 OPERABLE channels required.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All Reactor Coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required Reactor Coolant loops in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENT

4.4.1.1 The above required Reactor Coolant loops shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the Reactor Coolant loops listed below shall be OPERABLE and at least one of these Reactor Coolant loops shall be in operation.*

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

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

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

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 all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required Reactor Coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication at least once per 12 hours.

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

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the Reactor Coolant and/or residual heat removal (RHR) loops listed below shall be OPERABLE and at least one of these Reactor Coolant and/or RHR loops shall be in operation.**

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

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

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

d. Residual Heat Removal Loop A,

e. Residual Heat Removal Loop B,

APPLICABILITY: MODE 4

ACTION:

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

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

*A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless 1) the pressurizer water volume is less than 1288 cubic feet and/or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

**All Reactor Coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required Reactor Coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication at least once per 12 hours.

4.4.1.3.3 At least one Reactor Coolant or RHR loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.3.4 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water at least once per 31 days.*

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

COLD SHUTDOWN – LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE and/or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.4.1.3 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water at least once per 31 days.

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

A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless 1) the pressurizer water volume is less than 1288 cubic feet and/or 2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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

REACTOR COOLANT SYSTEM

COLD SHUTDOWN – LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with Reactor Coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.4.2.2 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water at least once per 31 days.

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

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

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.1 The Surveillance Requirements of Specification 4.0.5 shall be met, or; the pressurizer code safety valve shall have its lift set pressure verified under cold conditions.

Correction letter of 2-14-91

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.#

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

#Mode 3 applicability is exempted under the following conditions:

1. There has been a least 5 days of operation in MODES 5 or 6 since the reactor was last critical, and

2. All RCCAs are fully inserted with all CRDMs de-energized.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1288 cubic feet, (92% of indicated span) and at least two groups of pressurizer heaters each having a capacity of at least 125 kw.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable and capable of being manually cycled, within 1 hour:
   1) restore the PORV(s) to OPERABLE status or
   2) close the associated block valve(s) and maintain power to the block valve;

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

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour:
   1) restore the PORV to OPERABLE status or to a condition where it may be manually cycled* or
   2) close its associated block valve and remove power from the block valve;

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

c. With two PORVs inoperable and not capable of being manually cycled,
   1) within 1 hour:
      a) restore the PORVs to OPERABLE status or to a condition where they are capable of being manually cycled* or
      b) close the associated block valves and remove power from the block valves and
   2) within the next 72 hours:
      a) restore a minimum of two PORVs to OPERABLE status or
      b) restore a minimum of two PORVs to a condition where they are capable of being manually cycled*;

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

* If a PORV is inoperable but capable of being manually cycled, the associated block valve must be closed with power maintained to the block valve.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

d. With three PORVs inoperable and not capable of being manually cycled,
   1) within 1 hour:
      a) restore at least one PORV to OPERABLE status or to a condition where it is capable of being manually cycled*, and
      b) close and remove power from the block valves for any PORVs remaining inoperable and not capable of being manually cycled and
   2) within the next 72 hours:
      a) restore a minimum of two PORVs to OPERABLE status or
      b) restore a minimum of two PORVs to a condition where they can be manually cycled*;

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

e. With one block valve inoperable:
   1) within 1 hour:
      a) restore the block valve to OPERABLE status, or
      b) place the associated PORV in manual control and
   2) within the next 72 hours:
      a) restore the block valve to OPERABLE status or
      b) close the block valve and remove power from the block valve;

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

f. With two block valves inoperable:
   1) within 1 hour:
      a) restore the block valves to OPERABLE status, or
      b) place the associated PORVs in manual control and
   2) within 72 hours:
      a) restore at least two of the three block valves to OPERABLE status and
      b) ensure that the remaining inoperable block valve is closed and the power is removed;

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

* If a PORV is inoperable but capable of being manually cycled, the associated block valve must be closed with power maintained to the block valve.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

g. With three block valves inoperable:
   1) within 1 hour:
      a) restore the block valves to OPERABLE status, or
      b) place the associated PORVs in manual control and
   2) within the next 2 hours restore at least one of the three block valves to OPERABLE status and
   3) within the next 72 hours:
      a) restore at least two of the three block valves to OPERABLE status and
      b) ensure that the remaining inoperable block valve is closed and the power is removed;

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

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by.operating the valve through one complete cycle of full travel during MODES 3 or 4.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with the power removed in order to meet the requirements of 3.4.4.b, 3.4.4.c, or 3.4.4.d.

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

The ACTIONS may be entered separately for each steam generator tube.

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

The following pages were deleted:

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REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. One reactor building sump level,

b. One reactor building atmosphere radioactivity monitor (gaseous or particulate), and

c. One reactor building cooling unit condensate flow rate monitor.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the reactor building sump level monitor inoperable, perform surveillance requirement 4.4.6.2.1.d (Reactor Coolant System water inventory balance) at least once per 24 hours[(1)] and restore the required reactor building sump level 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.

b. With the required reactor building atmosphere radioactivity monitor inoperable, analyze grab samples of the containment atmosphere at least once per 24 hours or perform surveillance requirement 4.4.6.2.1.d (Reactor Coolant System water inventory balance) at least once per 24 hours[(1)] and restore the required reactor building atmosphere radioactivity monitor to OPERABLE status or verify the reactor building cooling unit condensate flow rate monitor is OPERABLE within 30 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the reactor building cooling unit condensate flow rate monitor inoperable, perform a CHANNEL CHECK of the required reactor building atmosphere radioactivity monitor at least once per 8 hours or perform surveillance requirement 4.4.6.2.1.d (Reactor Coolant System water inventory balance) at least once per 24 hours[(1)]; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With the reactor building sump level monitor and the reactor building cooling unit condensate flow rate monitor inoperable and with the reactor building atmosphere gaseous radioactivity monitor being the only remaining OPERABLE leakage

---

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

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

detection monitor, analyze grab samples of the containment atmosphere at least once per 12 hours and restore the required reactor building sump level monitor or the reactor building cooling unit condensate flow rate monitor to OPERABLE status within 7 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With the required reactor building atmosphere radioactivity monitor and the reactor building cooling unit condensate flow rate monitor inoperable, restore the required reactor building atmosphere radioactivity monitor or the reactor building air cooler condensate flow rate 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.

f. With all required monitoring systems inoperable, enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Reactor building atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Reactor building sump level-performance of CHANNEL CALIBRATION at least once per 18 months,

c. Reactor building atmosphere gaseous radioactivity monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION, AND ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified is Table 4.3-3,

d. Reactor building cooling unit condensate flow detector-performance of CHANNEL CALIBRATION at least once per 18 months.

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

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

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

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 33 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The leakage rate specified for each Reactor Coolant System Pressure Isolation Valve in Table 3.4-1 at a Reactor Coolant System pressure of 2235 ± 20 psig.

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

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

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the reactor building atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the reactor building sump inventory at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.(1) This requirement is not applicable to primary-to-secondary leakage.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

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

a. During startup following each refueling outage.

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

c. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve for valves denoted on Table 3.4-1 by an asterisk*.

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

4.4.6.2.3 Primary-to-secondary leakage shall be verified ≤ 150 gallons per day through any one steam generator at least once per 72 hours.(1)

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NO.	DESCRIPTION	NOMINAL SIZE (Inches)	ALLOWABLE LEAKAGE PER VALVE (Gallons per Minute)
8993 A, B, C	SI to Hot Legs	6	3
8992 A, B, C	SI High Head to Hot Legs	2	1
8990 A, B, C	SI High Head to Hot Legs	2	1
8988 A, B	SI Low Head to Hot Legs	6	3
8997 A, B, C	Primary SI High Head to Cold Legs	2	1
8995 A, B, C	Alternate SI High Head to Cold Legs	2	1
8998 A, B, C	SI to Cold Legs	6	3
8973 A, B, C	RHR Low Head to Cold Legs	6	3
*8948 A, B, C	Accumulators to Cold Legs	12	5
*8956 A, B, C	Accumulators to Cold Legs	12	5
8701 A, B	RHR Suction from Hot Legs	12	5
8702 A, B	RHR Suction from Hot Legs	12	5
8974 A, B	RHR Low Head to Cold Legs	10	5

*See Specification 4.4.6.2.2.c

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

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

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

At All Other Times:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.4-2

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

*Limit not applicable with $T_{avg} \leq 250°F$.

TABLE 4.4-3

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

*Not required with $T_{avg} \leq 250°F$

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1.0 microcurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries per gram.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

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

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

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

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

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TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	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.

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

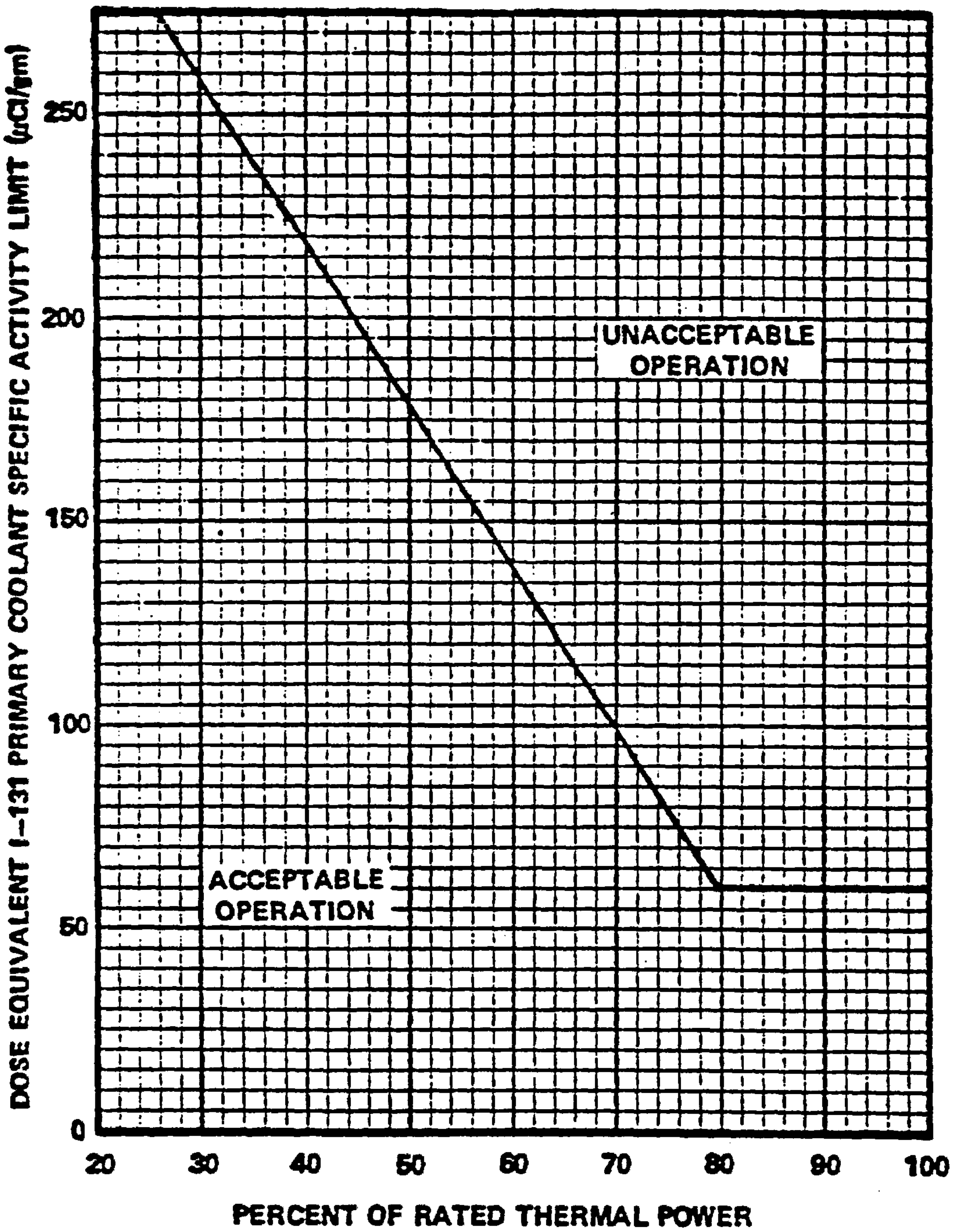

FIGURE 3.4-1

DOSE EQUIVALENT I—131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 µCi/gram Dose Equivalent I-131

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

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MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE A9154-1

LIMITING ART VALUES @ 56 EFPY: 1/4T, 153°F
3/4T, 138°F

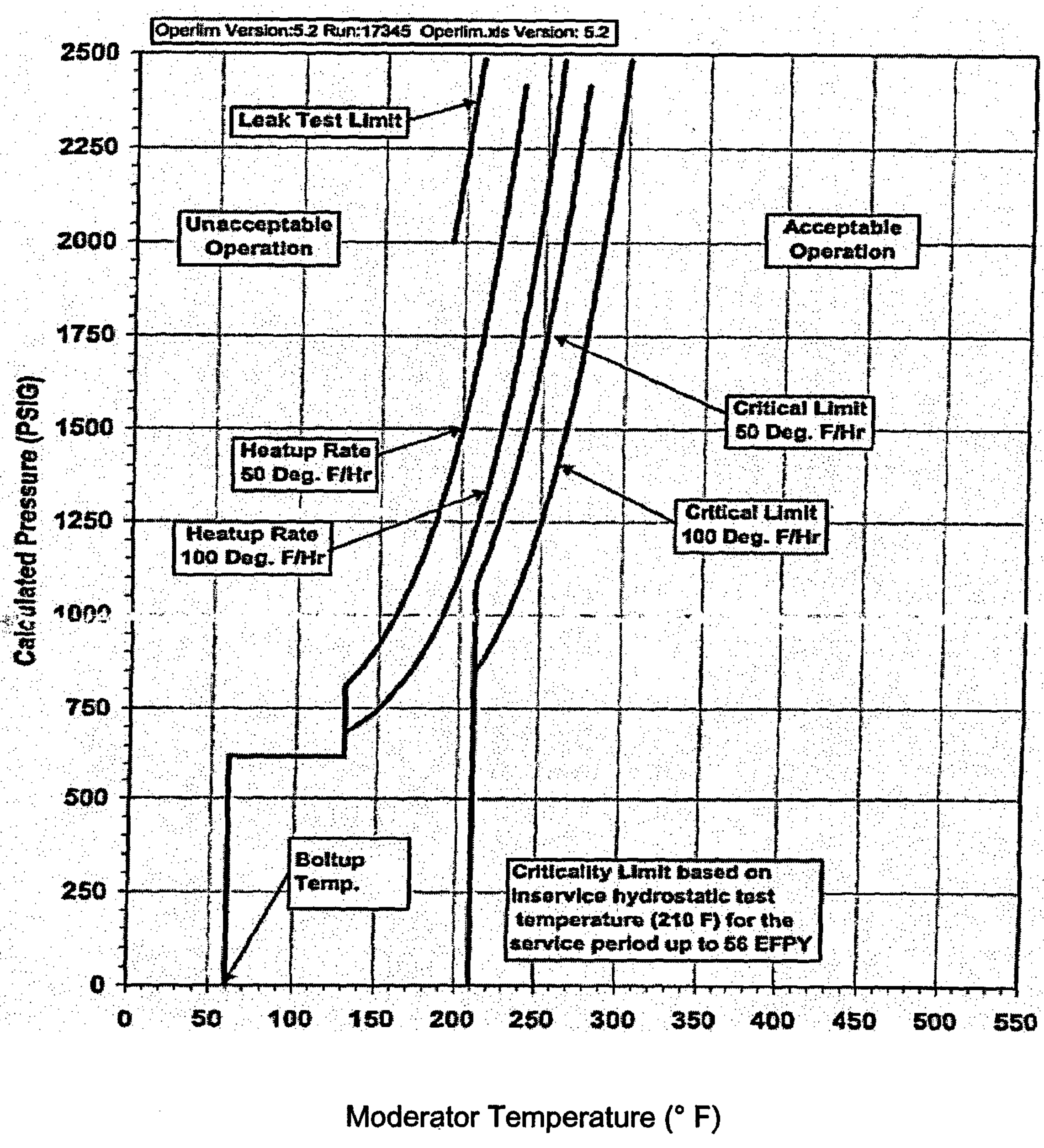

FIGURE 3.4-2 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 50 and 100°F/hr) Applicable for 56 EFPY (Without Margins for Instrumentation Errors) Using 1998 Appendix G Methodology

REACTOR COOLANT SYSTEM

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE A9154-1

LIMITING ART VALUES @ 56 EFPY: 1/4T, 153°F

3/4T, 138°F

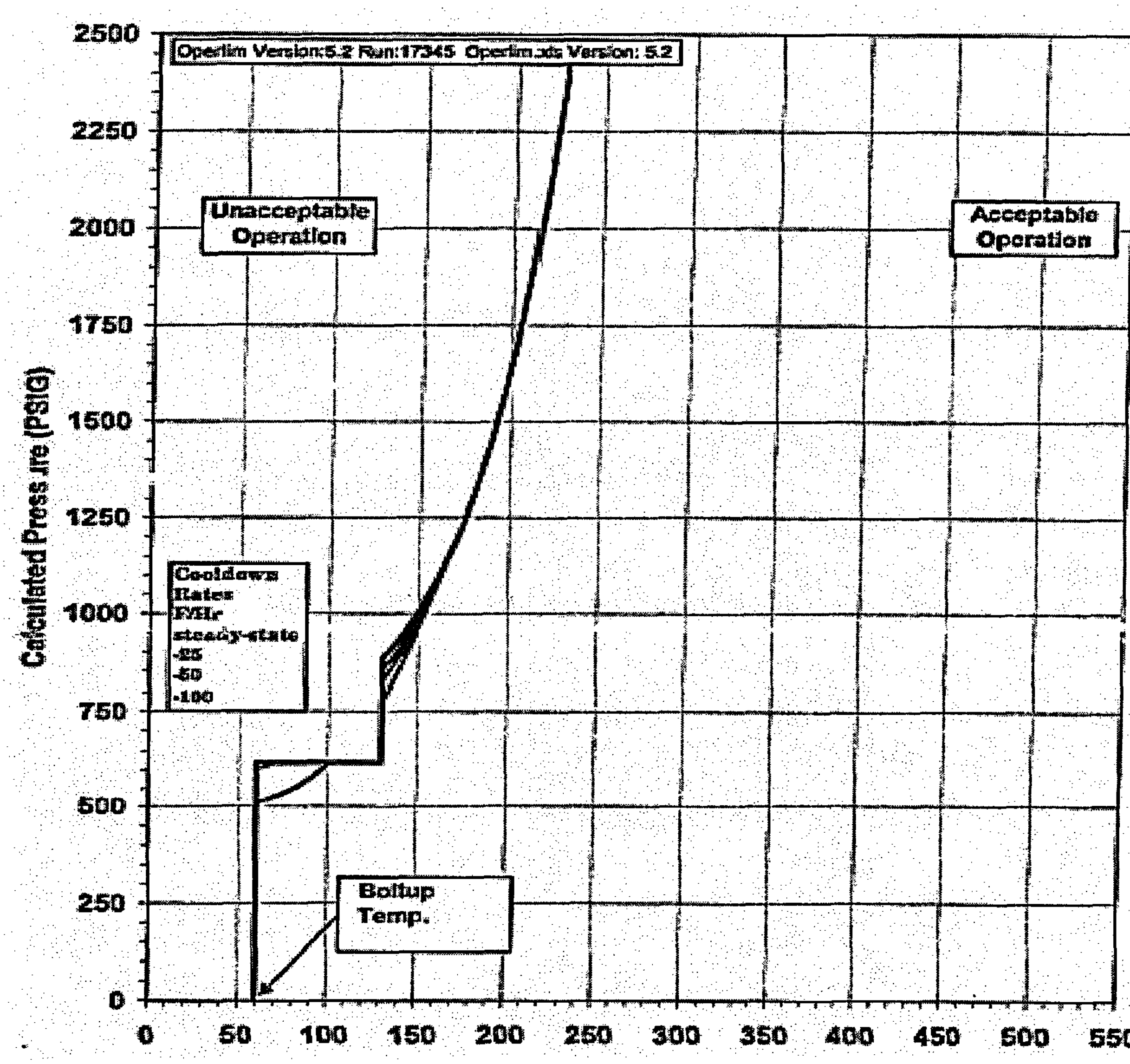

FIGURE 3.4-3 V. C. Summer Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 56 EFPY (Without Margins for Instrumentation Errors) Using 1998 Appendix G Methodology

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 auxiliary spray water temperature differential of 625°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the 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 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two RHR relief valves with:

1. A lift setting of less than or equal to 450 psig, and

2. The associated RHR relief valve isolation valves open; or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.7 square inches.

APPLICABILITY:

MODE 4 when the temperature of any RCS cold leg is less than or equal to 300°F, MODE 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 72 hours or depressurize and vent the RCS through a greater than or equal to 2.7 square inch vent within the next 8 hours.

b. With both RHR relief valves inoperable, within 8 hours either:

1. Restore at least one RHR relief valve to OPERABLE status, or

2. Depressurize and vent the RCS through a greater than or equal to 2.7 square inch vent.

c. In the event an RHR relief valve or RCS vent is 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 RHR relief valves or vent on the transient and any corrective action necessary to prevent recurrence.

d. In the event that two or more charging pumps are capable of injecting into the RCS, immediately initiate action to ensure a maximum of one charging pump is capable of injecting into the RCS#.

e. The provisions of Specification 3.0.4 are not applicable.

Two charging pumps may be capable of injecting into the RCS while swapping pumps, ≤15 minutes.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each RHR relief valve shall be demonstrated OPERABLE by:

a. Verifying the RHR relief valve isolation valves (8701A, 8701B, 8702A, and 8702B) are open at least once per 72 hours when the RHR relief valve is being used for overpressure protection.

b. Testing pursuant to Specification 4.0.5.

c. Verification of the RHR relief valve setpoint of at least one RHR relief valve, at least once per 18 months on a rotating basis.

4.4.9.3.2 The RCS vent shall be verified to be open at least once per 12 hours* when the vent is being used for overpressure protection.

4.4.9.3.3 At least two charging pumps shall be verified incapable of injecting into the RCS at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, verify these valves open at least once per 31 days.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 The requirements of Specification 4.0.5 are applicable.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7489 and 7685 gallons,

c. A boron concentration of between 2200 and 2500 ppm, and

d. A nitrogen cover-pressure of between 600 and 656 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within one hour and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2. Verifying that each accumulator isolation valve is open.

---

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that the isolation valve operator breaker opened at the motor control center and locked in the open position.

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

   1. When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) setpoint,

   2. Upon receipt of a safety injection test signal.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the residual heat removal sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours* or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

* The allowable outage time for each RHR train may be extended to 7 days for the purpose of maintenance and modification. This exception may only be used one time per RHR train and is not valid after December 31, 1997.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours 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.	8884	HHSI Hot Leg Injection	Closed
2.	8886	HHSI Hot Leg Injection	Closed
3.	8888A	LHSI Cold Leg Injection	Open
4.	8888B	LHSI Cold Leg Injection	Open
5.	8889	LHSI Hot Leg Injection	Closed
6.	8701A	RHR Inlet	Closed
7.	8701B	RHR Inlet	Closed
8.	8702A	RHR Inlet	Closed
9.	8702B	RHR Inlet	Closed
10.	8133A	Charging/HHSI Cross-Connect	Open
11.	8133B	Charging/HHSI Cross-Connect	Open
12.	8106	Charging Mini-Flow Header Isolation	Open

b. At least once per 31 days by:

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

2. Verify ECCS 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 reactor building which could be transported to the RHR and Spray Recirculation sumps and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

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

2. Of the areas affected with the reactor building at the completion of each reactor building entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection actuation and containment sump recirculation test signal.

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

a) Centrifugal charging pump

b) Residual heat removal pump

f. By verifying each ECCS pump's developed head at the test flow point for that pump is greater than or equal to the required developed head in accordance with Specification 4.0.5.

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

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

2. At least once per 18 months.

HPSI System
Valve Number

a. 8996A
b. 8996B
c. 8996C
d. 8994A
e. 8994B
f. 8994C
g. 8989A
h. 8989B
i. 8989C
j. 8991A
k. 8991B
l. 8991C

SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running and with recirculation flow:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 338 gpm, and

b) The total pump flow rate is less than or equal to 688 gpm.

i. By performing a flow test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For residual heat removal pump lines, with a single pump running the sum of the injection line flow rates is greater than or equal to 3663 gpm.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

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

A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

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 All charging pumps except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F by verifying that the motor circuit breakers have been secured in the open position.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 453,800 gallons,

b. A boron concentration of between 2300 and 2500 ppm of boron, and

c. A minimum water temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage 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 RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F.

CONTAINMENT SYSTEMS

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 measured overall integrated containment leakage rate exceeding 1.0 L_a, within 1 hour initiate action to be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the overall integrated leakage rate to less than or equal to 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than or equal to 0.60 L_a 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 intervals derived through the Containment Leakage Rate Testing Program and shall be determined per the program criteria.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.4.

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

c. Deleted.

---

* 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.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each reactor building air lock shall be OPERABLE with 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.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one reactor building air lock door inoperable:

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

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

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

   4. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each reactor building air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program.

b. Deleted.

c. At least once per six months by verifying that only one door in each air lock can be opened at a time.

d. Deleted.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Reactor building internal pressure shall be maintained between -0.1 and 1.5 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 COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The reactor building internal pressure shall be determined to be within the limits at least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at or above the following locations and shall be determined at least once per 24 hours:

a. Elevation 412' - 3 locations

b. Elevation 436' - 3 locations

c. Elevation 463' - 3 locations

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be OPERABLE.

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

ACTION:

If the structural integrity of the containment is found to be inoperable, restore the containment to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the containment shall be demonstrated in accordance with the Containment Inservice Inspection Program.

4.6.1.6.2 Deleted

4.6.1.6.3 In accordance with the Containment Leakage Rate Testing Program, the structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined by a visual inspection of these surfaces and verifying that no abnormal material or structural behavior is evident.

SURVEILLANCE REQUIREMENTS (Continued)

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LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 36-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 6-inch containment purge supply and exhaust isolation valves may be open for less than or equal to 1000 hours per 365 days.

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed close, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a 6-inch containment purge supply and/or exhaust isolation valve(s) open for more than 1000 hours per 365 days, close the open 6-inch valve(s) or isolate the penetration within 4 hours otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3, restore the inoperable valve(s) to OPERABLE status within 24 hours; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment purge supply and exhaust isolation valve shall be verified to be:

a. Closed at least once per 24 hours.

b. Sealed closed at least once per 31 days.

4.6.1.7.2 The cumulative time that the 6-inch purge supply and exhaust isolation valves have been open during the past 365 days shall be determined at least once per 7 days.

4.6.1.7.3 At least once per 30 months each containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

REACTOR BUILDING SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent reactor building spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and automatically transferring suction to the spray sump.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each reactor building spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

2. Verifying Containment Spray locations susceptible to gas accumulation are sufficiently filled with water.

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 195 psig when tested pursuant to Specification 4.0.5.

c. At least once per 18 months during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on each of the following test signals a Phase 'A', Reactor Building Spray Actuation, and Containment Sump Recirculation.

2. Verifying that each spray pump starts automatically on a Reactor Building Spray Actuation test signal.

d. At least once per 10 years by performing an air or smoke or equivalent flow test through each spray header and verifying each spray nozzle is unobstructed.

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

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

a. A spray additive tank containing a volume of between 3140 and 3230 gallons of between 20.0 and 22.0 percent by weight NaOH solution, and

b. A flow path capable of adding NaOH solution from the spray additive tank to the suction of each reactor building spray pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by:

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

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

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Phase 'A' signal.

d. At least once per 5 years by verifying each solution flow rate from the following drain connections in the spray additive system:

1. NaOH Tank to Loop A $\geq$ 15 gpm

2. NaOH Tank to Loop B $\geq$ 15 gpm

REACTOR BUILDING COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 Two independent groups of reactor building cooling units shall be OPERABLE with at least one of two cooling units OPERABLE in slow speed in each group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one group of the above required reactor building cooling units inoperable and both reactor building spray systems OPERABLE, restore the inoperable group of cooling units to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required reactor building cooling units inoperable, and both reactor building spray systems OPERABLE, restore at least one group of cooling units 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. Restore both above required groups of cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required reactor building cooling units inoperable and one reactor building spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of reactor building cooling units shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting each cooling unit group from the control room, and verifying that each cooling unit group operates for at least 15 minutes in the slow speed mode.

b. At least once per 18 months by:

   1. Verifying that each fan group starts automatically on a safety injection test signal.

   2. Verifying a cooling water flow rate of greater than or equal to 2,000 gpm to each cooling unit group.

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SURVEILLANCE REQUIREMENTS (Continued)

3. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a simulated SI test signal or on an ESFLS, as applicable.

4. At least once per 18 months, by verifying that each service water system booster pump starts automatically on a safety injection signal.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent groups of HEPA filter banks (associated with the OPERABLE reactor building cooling units of Specification 3.6.2.3) with at least one filter bank in each group, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one group of HEPA filter banks OPERABLE, restore one of the inoperable banks in the other group to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 The two groups of HEPA filter banks shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and verifying that the system operates for at least 15 minutes.

b. By performing required filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

c. At least once per 18 months by:

1. Verifying that the filter bypass damper can be opened by operator action.

2. Verifying that the filter bypass damper closes on a Safety Injection Test Signal.

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CONTAINMENT SYSTEMS

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

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

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

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

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

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

4.6.4.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE AT LEAST ONCE PER 18 MONTHS BY:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Reactor Building Purge and Exhaust isolation test signal, each Purge and Exhaust valve actuates to its isolation position.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

4.6.4.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

*All valves tested must have "as-left" lift setpoints that are within ±1% of the Lift Setting value listed in Table 3.7-2.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Each Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1(1)	58(1)
2	41
3	24

Notes: 1. With one inoperable safety valve in only one operating steam generator, the Maximum Allowable Power Range Neutron Flux High Setpoint may be increased to 81% of RATED THERMAL POWER provided the predicted Moderator Temperature Coefficient is negative (< 0 pcm/°F) at hot zero power assuming all rods out and no xenon.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

S/G A	S/G B	S/G C		Lift Setting *		Orifice Size
XVS-2806A	XVS-2806F	XVS-2806K		1176 psig ± 1%		4.515 In dia/16 sq in
XVS-2806B	XVS-2806G	XVS-2806L		1190 psig ± 3%		4.515 In dia/16 sq in
XVS-2806C	XVS-2806H	XVS-2806M		1205 psig ± 3%		4.515 In dia/16 sq in
XVS-2806D	XVS-2806I	XVS-2806N		1220 psig ± 3%		4.515 In dia/16 sq in
XVS-2806E	XVS-2806J	XVS-2806P		1235 psig ± 3%		4.515 In dia/16 sq in

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

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PLANT SYSTEMS

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Two motor-driven emergency feedwater pumps, each capable of being powered from separate emergency busses, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one emergency feedwater pump inoperable, restore the required emergency feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

c. With three emergency feedwater pumps inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

    1. Not used.

    2. Not used.

    3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that each automatic valve in the flow path from the condensate storage tank to the steam generators is in the fully open position whenever the emergency feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

5. Verifying that valves 1010-EF and 1007-EF are locked in the open position.

b. At least once per 3 months by verifying that the check valve in the instrument air supply line to the six emergency feedwater control valve air accumulators closes when the normal instrument air supply is not available.

c. At least once per 18 months during shutdown by verifying that:

1. Each emergency feed pump starts as designed automatically upon receipt of an emergency feedwater actuation test signal.

2. The six emergency feedwater control valves can be closed and held closed for three hours with air from the accumulators when the normal instrument air supply is not available.

3. The turbine driven emergency feedwater pump can be manually stopped from the main control board by closing the steam supply valve with air from the accumulator when the normal instrument air supply is not available.

4. Each automatic valve in the flow path actuates to its correct position on receipt of an emergency feedwater actuation test signal.

d. In accordance with the Inservice Testing Program as required by Specification 4.0.5 by verifying:

1. The developed head of each emergency feedwater pump at the flow test point is greater than or equal to the required developed head. Notes: 1) Not required to be performed for the turbine driven emergency feedwater pump until secondary steam supply pressure is greater than 865 psig. 2) The provisions of Specification 4.0.4 are not applicable for the turbine driven emergency feedwater pump.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained volume of at least 179,850 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The service water system shall be demonstrated OPERABLE at least once per 12 hours by verifying service water system pressure whenever the service water system is the supply source for the emergency feedwater pumps.

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours.
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.

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 but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours;

Otherwise, reduce power to less than or equal to 5 percent of RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 7 seconds when tested pursuant to Specification 4.0.5.

FEEDWATER ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Each feedwater isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

MODE 1 — With one feedwater isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 72 hours;

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

MODES 2 and 3 — With one feedwater isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each feedwater isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of the primary coolant and the steam generator shells shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the primary coolant or the steam generator shell is less than 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days 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.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.4 At least two service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days 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.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The service water pond (ultimate heat sink) shall be OPERABLE with:

a. A minimum water level at or above elevation 416.5 Mean Sea Level, USGS datum, and

b. A water temperature of less than or equal to 90.5°F at the discharge of the service water pumps.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.5 The service water pond shall be determined OPERABLE at least once per 24 hours by verifying the water temperature and water level to be within their limits.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS)

LIMITING CONDITION FOR OPERATION

3.7.6 Two CREFS trains shall be OPERABLE.*

APPLICABILITY: ALL MODES

ACTION:

a. MODES 1, 2, 3 and 4:

1. With one CREFS train inoperable for reasons other than ACTION 3.7.6.a.2, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one or more CREFS trains inoperable due to an inoperable control room envelope (CRE) boundary, immediately initiate action to implement mitigating actions and verify within 24 hours that the mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits and restore CRE boundary to OPERABLE status within 90 days. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With both CREFS trains inoperable for reasons other than ACTION 3.7.6.a.2, immediately enter LCO 3.0.3.

b. MODES 5 and 6:

1. With one CREFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable train to OPERABLE status within 7 days, or immediately place the OPERABLE CREFS train in the emergency mode of operation or immediately suspend movement of irradiated fuel assemblies.

2. With both CREFS trains inoperable or one or more CREFS trains inoperable due to an inoperable CRE boundary, immediately suspend movement of irradiated fuel assemblies.

3. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6 Each CREFS train shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 85°F.

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the CREFS train operates for at least 15 minutes.

c. By performing required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

* The control room envelope (CRE) boundary may be opened intermittently under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by verifying that on a simulated SI or high radiation test signal, each CREFS train automatically switches into an emergency mode of operation with flow through the HEPA filters and charcoal adsorber banks.

e. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Habitability Program.

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.7 All snubbers on systems required for safe shutdown/accident mitigation shall be OPERABLE. This includes safety and non-safety related snubbers on systems used to protect the code boundary and to ensure the structural integrity of these systems under dynamic loads.

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

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation for Specification 4.7.7 on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7 Each snubber shall be demonstrated OPERABLE by performance of the Snubber Testing Program in Section 6.8.4.n.

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3/4 7-17
3/4 7-18
3/4 7-19
3/4 7-20
3/4 7-21

PLANT SYSTEMS

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3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.8 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.8.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive materials:

   1. With a half-life greater than 30 days (excluding Hydrogen 3), and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.8.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

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Pages 3/4 7-26 through 3/4 7-36 have been deleted.

3/4.7.9 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.9 The temperature of each area shown in Table 3.7-7 shall be maintained below the limits indicated in Table 3.7-7.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-7:

a. For more than eight hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to below its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.9 The temperature in each of the areas of Table 3.7-7 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-7

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	Charging-SI Pump Room #1	102
2.	Charging-SI Pump Room #2 (swing)	102
3.	Charging-SI Pump Room #3	102
4.	RHR-Spray Pump Room #1	102
5.	RHR-Spray Pump Room #2	102
6.	MCC 1DA2Y Room	102
7.	Switchgear 1DB1 and MCC 1DB2Y Room	102
8.	Switchgear 1DA Room	102
9.	Switchgear 1DB Room	102
10.	Battery 1A Room	88
11.	Battery 1B Room	88
12.	Charger 1A Room	102
13.	Charger 1B Room	102
14.	Charger 1A/1B Room	102
15.	Relay Room	83
16.	Component Cooling pump "A" Speed Switch Room	102
17.	Component Cooling pump "B" Speed Switch Room	102
18.	Component Cooling Pump "C" Speed Switch Room	102
19.	Evacuation Panel "A" Room	83
20.	Evacuation Panel "B" Room	83
21.	Service Water Booster Pumps Area	102
22.	Emergency Feedwater Pumps Area	102
23.	Diesel Generator 1A Room	120
24.	Diesel Generator 1B Room	120
25.	Service Water Pump/Screen Room	118
26.	Service Water Switchgear Room "A"	102
27.	Service Water Switchgear Room "B"	102
28.	Service Water Switchgear Room "C"	102
29.	Diesel Generator Exciter Cabinet Room "A"	102
30.	Diesel Generator Exciter Cabinet Room "B"	102

3/4.7.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.7.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

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PLANT SYSTEMS

3/4.7.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.7.12 The combination of initial enrichment and cumulative burnup for spent fuel assemblies stored in Region 2 shall be within the acceptable domain of Figure 3.7-1.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

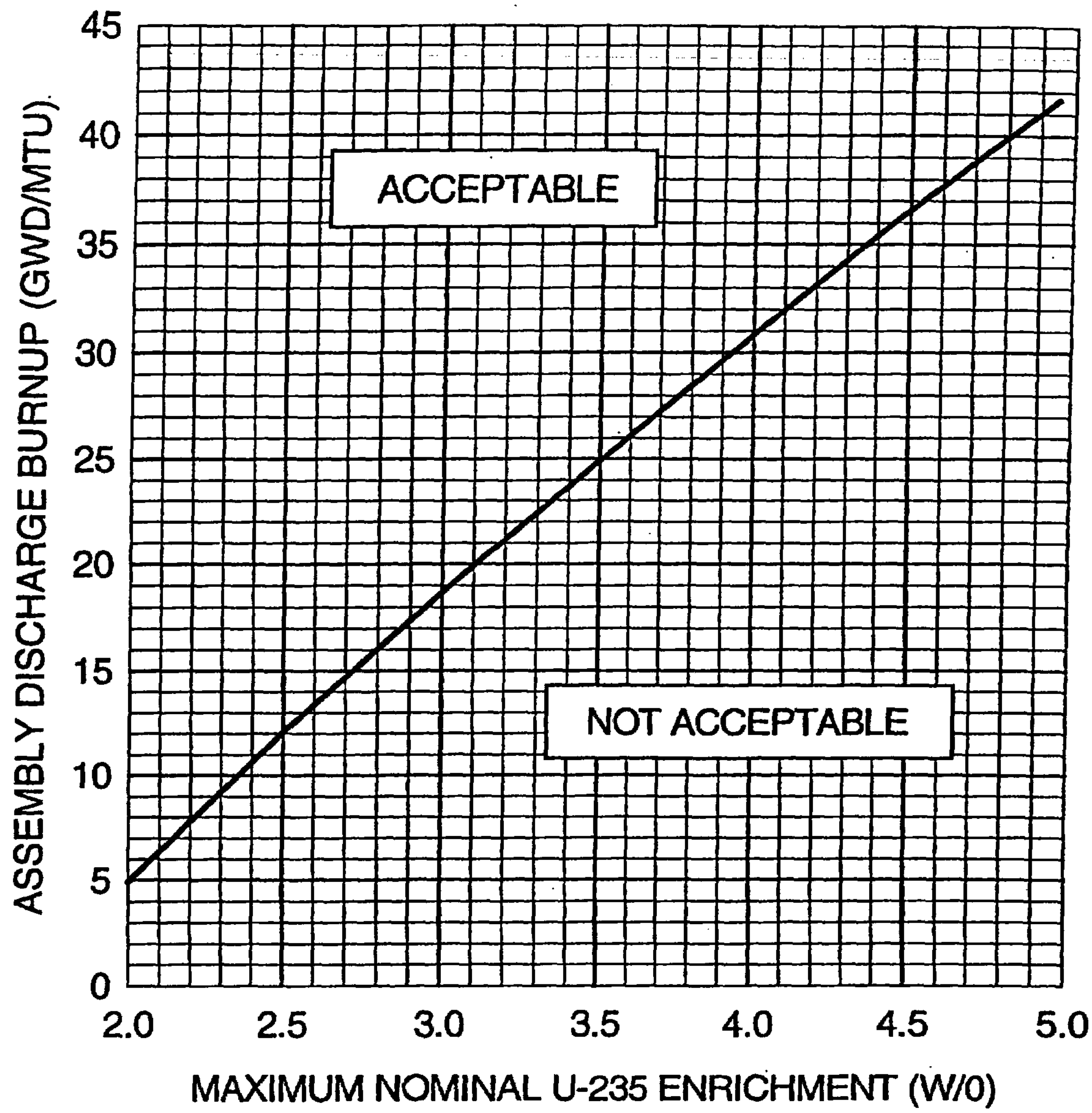
ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12 The burnup of each fuel assembly stored in Region 2 shall be ascertained by careful analysis of its burnup history prior to storage in Region 2. A complete record of such analysis shall be kept for the time period that the fuel assembly remains in Region 2 of the spent fuel pool.



Notes: 1. Fuel assemblies with enrichments less than 2.0 W/0 must meet the burn-up requirements of 2.0 W/0 U-235 assemblies.

2. Use of the following polynomial fit is acceptable, where E = Enrichment (W/0):

Assembly Discharge Burnup = $0.1246\ E^3 - 1.91\ E^2 + 20.9205\ E - 30.2482$

FIGURE 3.7-1 REQUIRED FUEL ASSEMBLY BURN-UP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.13 The boron concentration in the spent fuel pool, the fuel transfer canal, and the cask loading pit shall be maintained at a boron concentration greater than or equal to 500 ppm.

APPLICABILITY: Whenever new or irradiated fuel is being moved (non-refueling movement) in the spent fuel pool, fuel transfer canal, or cask loading pit.

ACTION:

With the requirements of the above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pool, the fuel transfer canal, and the cask loading pit until the boron concentration in the area where fuel is being moved shall be verified greater than or equal to 500 ppm.

SURVEILLANCE REQUIREMENTS

4.7.13 The boron concentration of the spent fuel pool, fuel transfer canal, or cask loading pit shall be determined by chemical analysis at least once per 72 hours when moving new or irradiated fuel in the spent fuel pool, transfer canal, or cask loading pit.

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 Emergency Diesel Generators (EDG), each with:

1. A separate day fuel tank containing a minimum volume of 360 gallons of fuel,

2. A separate fuel storage system containing a minimum volume of 48,500 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:

1. Demonstrate the OPERABILITY of the remaining offsite A.C. sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter, and

2. If either EDG has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.3 separately for each such EDG within 24 hours unless the diesel is already operating, and

3. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one EDG of 3.8.1.1.b inoperable:

1. Demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter, and

2. *If the EDG became inoperable due to any cause other than preplanned preventive maintenance or testing:

a) determine the OPERABLE EDG is not inoperable due to a common cause failure within 24 hours, or

b) demonstrate the OPERABILITY of the remaining EDG by performing Surveillance Requirement 4.8.1.1.2.a.3 within 24 hours,

and

* Completion of Action b.2 is required regardless of when the inoperable EDG is restored to OPERABILITY.

ACTION: (Continued)

3. Within 4 hours, verify that required systems, subsystems, trains, components and devices that depend on the remaining EDG as a source of emergency power are also OPERABLE and in MODE 1, 2, or 3, that the Turbine Driven Emergency Feed Pump is OPERABLE. If these conditions are not satisfied within 4 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. Restore the EDG to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, unless the following condition exists:

   a) The requirement for restoration of the EDG to OPERABLE status within 72 hours may be extended to 14 days if the Alternate AC (AAC) power source is or will be available within 1 hour, as specified in the Bases, and

   b) If at any time the AAC availability cannot be met, either restore the AAC to available status within the remainder of the 72 hours in 4.a (not to exceed 14 days from the time the EDG originally became inoperable), or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 30 hours.

c. With one offsite circuit and one EDG inoperable:

1. Demonstrate the OPERABILITY of the remaining offsite A.C. source by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter, and

2. *If the EDG became inoperable due to any cause other than preplanned preventative maintenance or testing:

   a) determine the OPERABLE EDG is not inoperable due to a common cause failure within 8 hours, or

   b) demonstrate the OPERABILITY of the remaining EDG by performing Surveillance Requirement 4.8.1.1.2.a.3 within 8 hours,

   and

3. Within 2 hours, verify that required systems, subsystems, trains, components and devices that depend on the remaining EDG as a source of emergency power are also OPERABLE and in MODE 1, 2, or 3, that the Turbine Driven Emergency Feed Pump is OPERABLE. If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. Restore one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

5. 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 Statement a. or b., as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source.

* Completion of Action c.2 is required regardless of when the inoperable EDG is restored to OPERABILITY.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With two of the required offsite A. C. circuits inoperable:

1. Demonstrate the OPERABILITY of the two EDG's by sequentially performing Surveillance Requirement 4.8.1.1.2.a.3 on both within 8 hours, unless the EDG's are already operating, and

2. Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours.

3. Following restoration of one offsite source, follow Action Statement a. with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit.

e. With two of the above required EDG's inoperable:

1. Demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter, and

2. Restore one of the inoperable EDG's to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. Following restoration of one EDG, 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.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indication of power availability for each Class 1E bus and its preferred offsite power source.

4.8.1.1.2 Each EDG shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank and fuel storage tank.

2. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

3. Verifying the diesel generator can start* and accelerate to synchronous speed (504 rpm) with generator voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz.

4. Verifying the generator is synchronized, gradually loaded* to an indicated 4150-4250 kW** and operates for at least 60 minutes.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by removing accumulated water from the day tank.

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks.

d. By sampling new fuel oil based on the applicable ASTM standard prior to addition to storage tanks and:

1. By verifying based on the tests specified in the applicable ASTM standard prior to addition to the storage tanks that the sample has:

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

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

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

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

d) A clear and bright appearance when tested based on the applicable ASTM standard.

2. By verifying within 30 days of obtaining the sample that the specified properties are met when tested based on the applicable ASTM standard.

e. At least once every 31 days by obtaining a sample of fuel oil based on the applicable ASTM standard, and verifying that total contamination is less than 10 mg/liter when checked based on the applicable ASTM standard.

f. At least once per 184 days by:

1. Verify each EDG starts from standby conditions and:
   a) In less than or equal to 10 seconds, achieves a voltage greater than 6480 volts (7200 - 720 volts) and a frequency greater than 58.8 Hz (60 - 1.2 Hz).
   b) Achieve a steady state voltage greater than 6480 volts but less than 7920 volts and a steady state frequency greater than 58.8 Hz but less than 61.2 Hz.

   The EDG shall be started for this test by using one of the following signals:
   a) Simulated loss of offsite power by itself.
   b) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
   c) An ESF actuation test signal by itself.
   d) Simulated degraded offsite power by itself.
   e) Manual.

2. The generator shall be manually synchronized, loaded to an indicated 4150-4250 kW** in less than or equal to 60 seconds, and operate for at least 60 minutes.

g. At least once every 18 months by:

1. Deleted

2. Verifying that on rejection of a load of greater than or equal to 729 kW, the voltage and frequency are maintained at 7200 ± 720 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the generator capability to reject a load of 4250 kW without tripping. The generator voltage shall not exceed 7920 volts during and following the load rejection.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

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

    a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

    b) Verifying 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 of these loads, the steady-state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

5. Verifying that on an ESF actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. Verify that the EDG starts from standby conditions and in less than or equal to 10 seconds, achieves a voltage greater than 6480 volts and a frequency greater than 58.8 Hz. After steady state operation is obtained, the EDG shall be verified to have a voltage greater than 6480 volts but less than 7920 volts and a frequency greater than 58.8 Hz but less than 61.2 Hz. After 5 minutes of standby operation verify that on a simulated loss of offsite power:

    a) the loads are shed from the emergency busses,

    b) the diesel generator does not connect to the bus for at least 5 seconds, and

    c) that subsequent loading of the diesel generator is in accordance with design requirements.

6. Simulating a loss of offsite power in conjunction with an ESF actuation test signal, and

    a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

    b) Verifying the EDG starts in the emergency mode, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes and maintains the steady state voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz.

    c) Verifying that all EDG trips, except engine overspeed, generator differential and low lube oil pressure are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

7. Verifying the EDG operates for at least 24 hours:

   a) The EDG shall be loaded to the continuous rating (4150-4250 kW**) for the time required to reach engine temperature equilibrium, at which time the EDG shall be loaded to an indicated target value of 4676 kW (between 4600-4700 kW**) and maintained for 2 hours.

   b) During the remaining 22 hours of this test, the EDG shall be loaded to an indicated 4150-4250 kW**.

   c) During this test the steady state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

8. Verifying that the auto-connected loads to each EDG do not exceed the 2000 hour rating of 4548 kW.

9. Verifying the EDG's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

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

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) Barring Device
    b) Remote-Local-Maintenance Switch

14. Verifying that within 5 minutes of operating the diesel generator for at least 1 hour at a load of 4150-4250 kW** the diesel starts on the auto-start signal (Loss of Off-Site Power signal), energizes the emergency busses with permanently connected loads

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

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 of these loads, the steady-state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

h. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 504 rpm in less than or equal to 10 seconds.

i. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III subsection ND of the ASME Code in accordance with Specification 4.0.5.

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(Table 4.8-1 was deleted)

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator* with:

1. A day fuel tank containing a minimum volume of 360 gallons of fuel,

2. A fuel storage system containing a minimum volume of 42,500 gallons of fuel, and

3. A fuel transfer pump,

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS positive reactivity changes, movement of irradiated fuel, or crane operation with loads over 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 reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 (with the exception of 4.8.1.1.2.a.4).

* ESF load sequencer may be deenergized in Modes 5 and 6 provided that the loss of voltage and degraded voltage relays are disabled.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery bank No. 1A and its associated full capacity charger.

b. 125-volt Battery bank No. 1B and its associated full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one 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

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days 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.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of 10 of the connected cells is $\geq$ 60°F.

c. At least once per 18 months 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. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

4. The battery charger will supply at least 300 amperes at 132 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, 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 4.8.2.1.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 REQUIREMENTS

Parameter	CATEGORY A[1]	CATEGORY B[2]	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	≥ 1.200[b]	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[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) Any Category B parameter not within its allowable value indicates an inoperable battery.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt battery bank and its associated full capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; and initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible.

b. With the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one 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

4.8.2.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open between redundant busses:

a. Train A A.C. Emergency Busses consisting of:
   1. 7200 volt Emergency Busses # 1DA and 1EA.
   2. 480 volt Emergency Busses # 1DA1, 1DA2 and 1EA1.
b. Train B A.C. Emergency Busses consisting of:
   1. 7200 volt Emergency Busses # 1DB and 1EB.
   2. 480 volt Emergency Busses # 1DB1, 1DB2, and 1EB1.
c. 120 volt A.C. Vital Busses # 5902 and 5901 energized from an associated inverter connected to D.C. Bus # 1HA*.
d. 120 volt A.C. Vital Busses # 5904 and 5903 energized from an associated inverter connected to D.C. Bus # 1HB*.
e. 120 volt A.C. Vital Bus #5907 energized.
f. 120 volt A.C. Vital Bus #5908 energized.
g. 125 volt D.C. Bus 1HA energized from Battery Bank #1A.
h. 125 volt D.C. Bus 1HB energized from Battery Bank #1B.

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

ACTION:

a. With one of the required trains of A.C. Emergency busses not fully energized, re-energize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Vital Bus not energized, re-energize the A.C. Vital Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one of A.C. Vital Busses #5901, 5902, 5903, or 5904 either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus re-energize the A.C. Vital Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*The inverters may be disconnected from their D.C. Bus for up to 24 hours as necessary for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. Bus.

ACTION: (Continued)

d. With one D.C. bus not energized from its associated Battery Bank, re-energize the D.C. bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. Emergency Busses consisting of two 7200 volt and three 480 volt A.C. Emergency Busses.

b. Three 120 volt A.C. Vital Busses energized from their associated inverters connected to their respective D.C. Busses.

c. One 125 volt D.C. Bus energized from its associated battery bank.

APPLICABITLITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, and initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 For each containment penetration provided with a penetration conductor overcurrent protective device(s), each device(s) shall be operable.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 Protective devices required to be operable as containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE.

a. At least once per 18 months:

   1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

      (a) A CHANNEL CALIBRATION of the associated protective relays, and

      (b) An integrated system functional test which includes simulat automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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ELECTRICAL POWER SYSTEMS

CIRCUIT PROTECTION DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.3 Circuit breakers for non-Class 1E cables located in trays which do not have cable tray covers and which provide protection for cables that if faulted could cause failure in both adjacent, redundant Class 1E cables shall be OPERABLE.

APPLICABILITY: All modes

ACTION:

a. With one or more of the above required non-Class 1E circuit breaker(s) inoperable, within 72 hours, either:

   1. Restore the circuit breaker(s) to OPERABLE status; or
   2. De-energize the circuit breaker(s); or
   3. Establish a one (1) hour roving fire watch for those areas in which redundant systems or components could be damaged.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above required circuit breakers shall be demonstrated OPERABLE.

a. At least once per eighteen (18) months:

   1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing the following:

      (a) A CHANNEL CALIBRATION of the associated protective relays, and

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

      (c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least ten percent (10%) of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breaker's nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least ten percent (10%) of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per sixty (60) months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

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

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6* with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

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 greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 The following valves shall be verified locked closed** at least once per 72 hours: 8430, 8454, 8441 and 8439.

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**Valves may be opened under administrative control to add borated makeup.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the reactor coolant system at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

REFUELING OPERATIONS

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical a period of time within the acceptable domain of Figure 3.9-1, but not less than 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

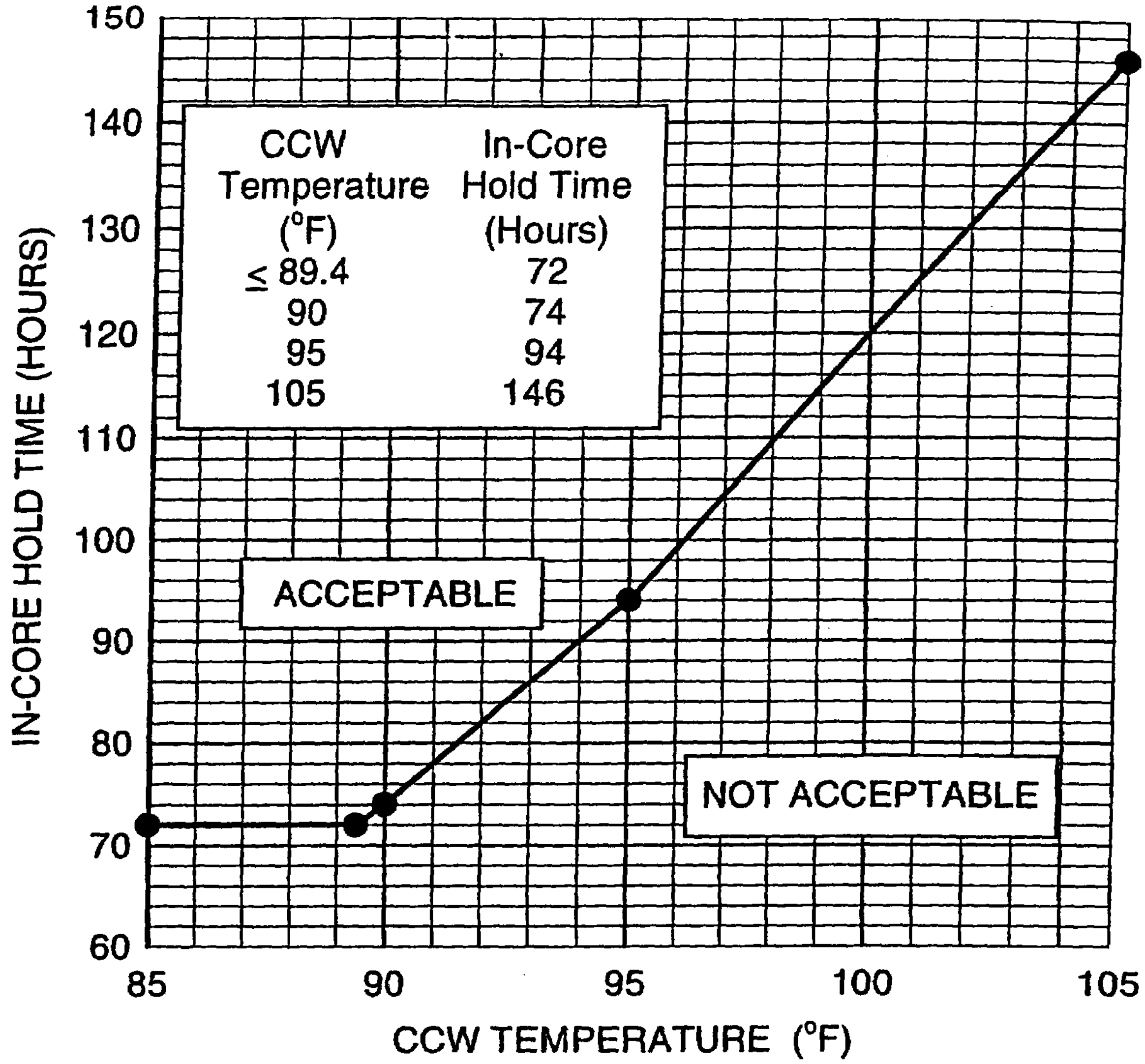
ACTION:

With the reactor subcritical for less than 72 hours, immediately suspend all movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for greater than 72 hours but not within the acceptable domain of Figure 3.9-1, immediately suspend movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3.1 The reactor shall be determined to have been subcritical for a period of time within the acceptable domain of Figure 3.9-1 by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.2 Prior to moving irradiated fuel from the reactor pressure vessel, and at least once every 12 hours during movement of irradiated fuel, verify the CCW temperature at the inlet to the Spent Fuel Pool Cooling System heat exchanger is within the acceptable domain of Figure 3.9-1.



Note: The use of linear interpolation between CCW temperatures reported above is acceptable to determine the minimum incore hold time.

FIGURE 3.9-1 REQUIRED IN-CORE HOLD TIME AS A FUNCTION OF COMPONENT COOLING WATER (CCW) TEMPERATURE

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3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1. A minimum capacity of 3250 pounds, and

   2. An overload cut off limit less than or equal to 2700 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

   1. A minimum capacity of 3000 pounds, and

   2. A load indicator which shall be used to prevent lifting loads in excess of 1000 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2700 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 1250 pounds.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no residual heat removal loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.1.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

4.9.7.1.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water at least once per 31 days.

* The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.2.1 At least one residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

4.9.7.2.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water at least once per 31 days.

* Prior to initial criticality the residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

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3/4.9.9 WATER LEVEL - REFUELING CAVITY AND FUEL TRANSFER CANAL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel or the refueling cavity when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s),

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of the below listed Specifications (a. and b.) shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Either Specifications 4.2.2.2 or 4.2.2.4 and Specification 4.2.2.5.

b. Specification 4.2.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of start up and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during start up and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating start up and PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (shutdown and control) rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the position indication system inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indication systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

---

* This requirement is not applicable during the initial calibration of the rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 Deleted by Amendment 104.
3.11.1.2 Deleted by Amendment 104.
3.11.1.3 Deleted by Amendment 104.

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Condensate Storage Tank
b. Outside Temporary Storage Tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1 Deleted by Amendment 104.
4.11.1.2 Deleted by Amendment 104.
4.11.1.3 Deleted by Amendment 104.

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

SETTLING POND

LIMITING CONDITION FOR OPERATION

3.11.1.5 The quantity of radioactive material contained in each settling pond shall be limited by the following expression:

$$\frac{264}{V} \cdot \sum_j \frac{A_j}{C_j} < 1.0$$

excluding tritium and dissolved or entrained noble gases, where,

A_j = Pond inventory limit for single radionuclide " j ", in curies.

C_j = 10 CFR 20, Appendix B, Table 2, column 2, concentration for single radionuclide " j ", microcuries/ml.

V = design volume of liquid and slurry in the pond, in gallons.

264 = Conversion unit, microcuries/curie per milliliter/gallon.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in the settling pond exceeding the above limit, immediately suspend all additions of radioactive material to the pond and within 48 hours reduce the pond contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.5 The quantity of radioactive material contained in each batch of slurry (used powdex resin) to be transferred to the settling ponds shall be determined to be within the above limit by analyzing a representative sample of the slurry, and batches to be transferred to the settling ponds shall be limited by the expression:

$$\sum_j \frac{Q_j}{C_j} < 1.0$$

SURVEILLANCE REQUIREMENTS (Continued)

where

Q_j = concentration of radioactive materials in wet, drained slurry (used powdex resin) for radionuclide " j " excluding tritium, dissolved or entrained noble gas and radionuclides with less than 8 day half-life, in microcuries per gram. The analysis shall include at least Ce-144, Cs-134, Cs-137, Sr-89, Sr-90, Co-58 and Co-60. Estimates of Sr-89, Sr-90, batch concentrations shall be based on the most recently available quarterly composite analyses.

C_j = 10 CFR 20, Appendix B, Table 2, column 2, concentration for single radionuclide " j ", in microcuries/milliliter.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.1 Deleted by Amendment 104.
3.11.2.2 Deleted by Amendment 104.
3.11.2.3 Deleted by Amendment 104.
3.11.2.4 Deleted by Amendment 104.

3.11.2.5 The concentration of oxygen in the waste gas holdup system 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 holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within 1 hour and less than or equal to 2% by volume within 48 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.1 Deleted by Amendment 104.
4.11.2.2 Deleted by Amendment 104.
4.11.2.3 Deleted by Amendment 104.
4.11.2.4 Deleted by Amendment 104.

4.11.2.5 The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

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 131,000 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 and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 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 at least one per 24 hours when radioactive materials are being added to the tank.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-3.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-4.

5.2 REACTOR BUILDING

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, pre-stressed, post-tensioned reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 126 feet.

b. Nominal inside height = 187 feet.

c. Minimum thickness of concrete walls = 4 feet.

d. Minimum thickness of concrete roof = 3 feet.

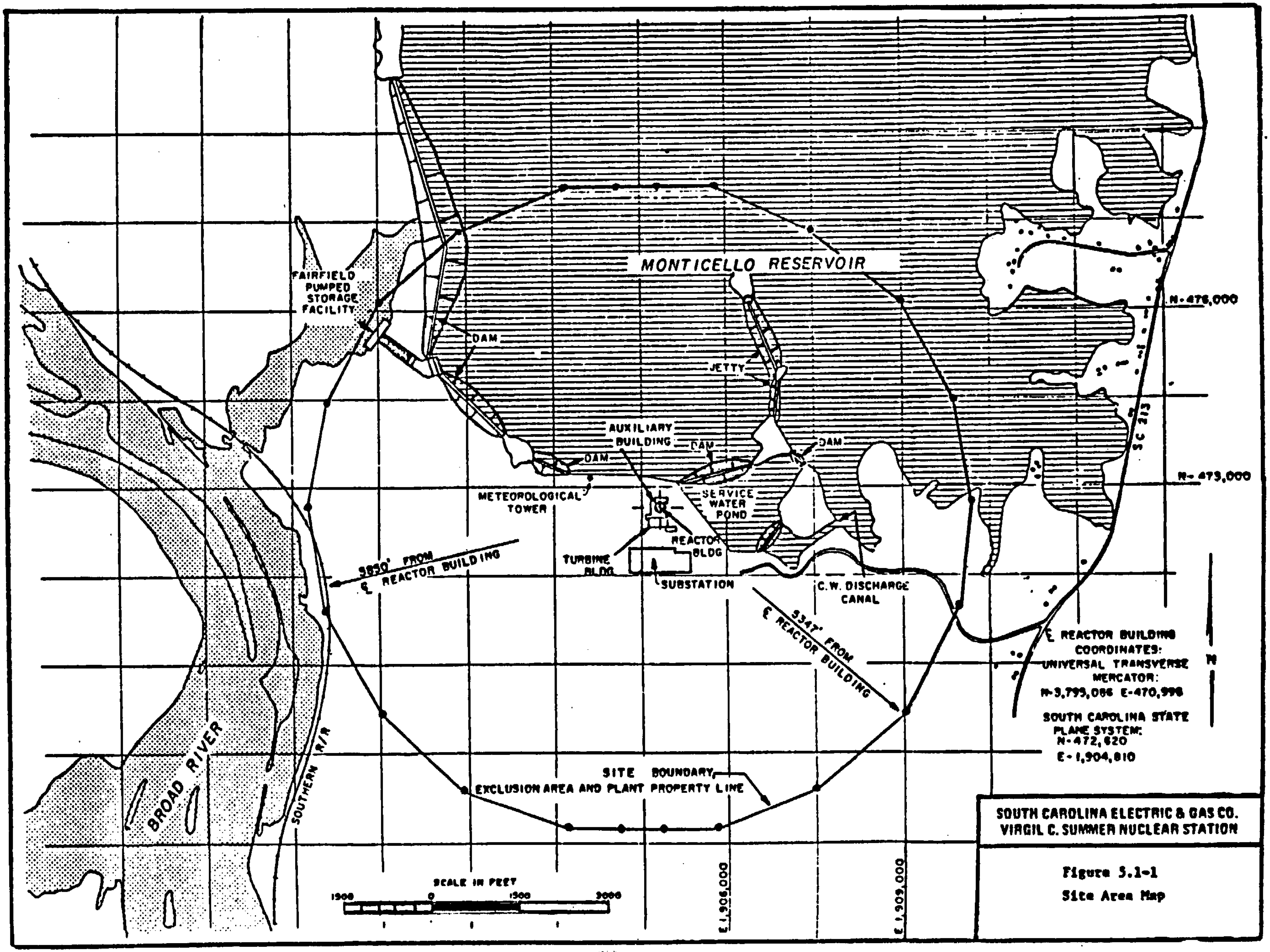
e. Minimum thickness of concrete floor pad = 4 feet.

f. Nominal thickness of steel liner = 0.25 inches.

g. Net free volume = 1.842×10^6 cubic feet.

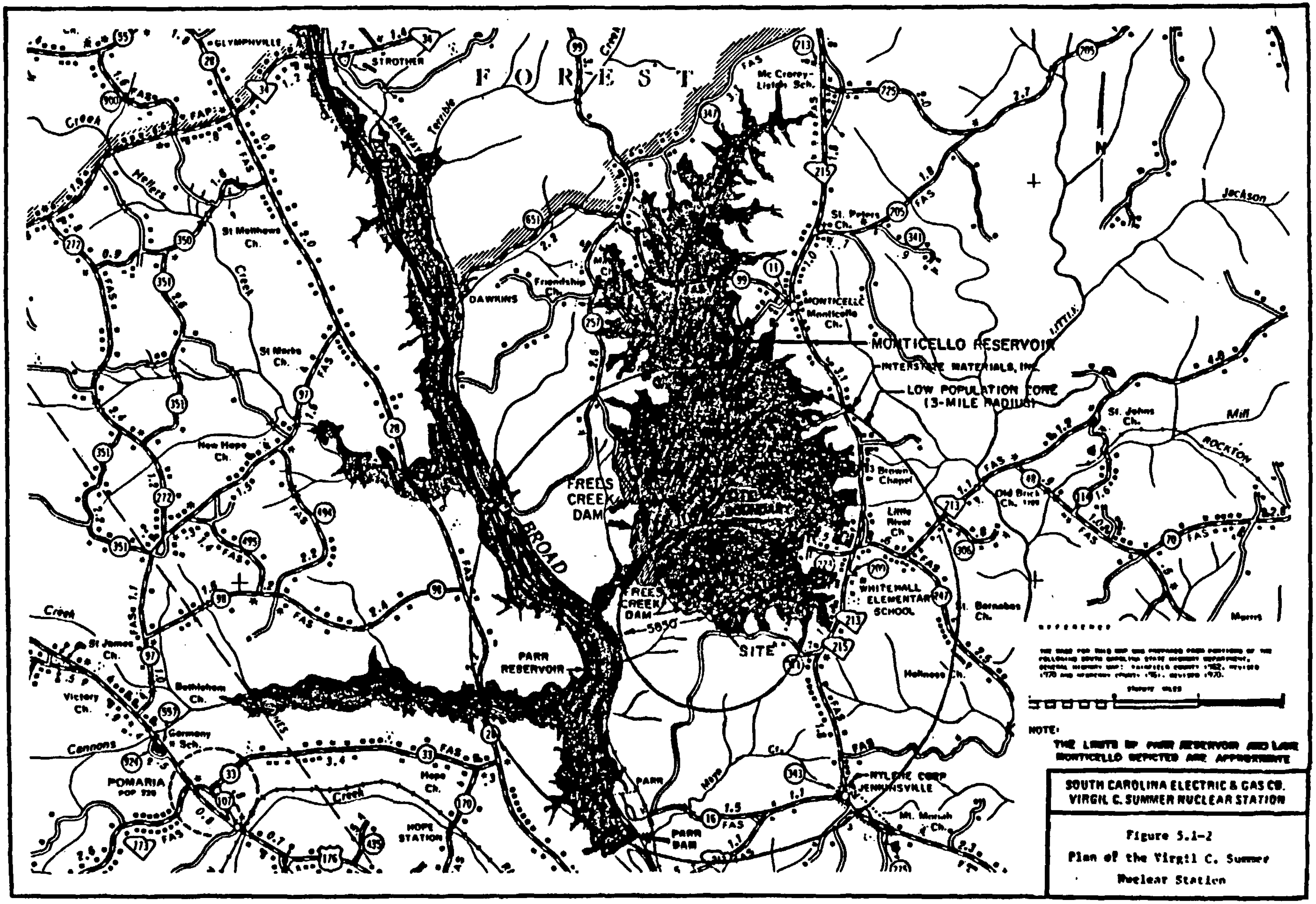
DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 57 psig and a temperature of 283°F.



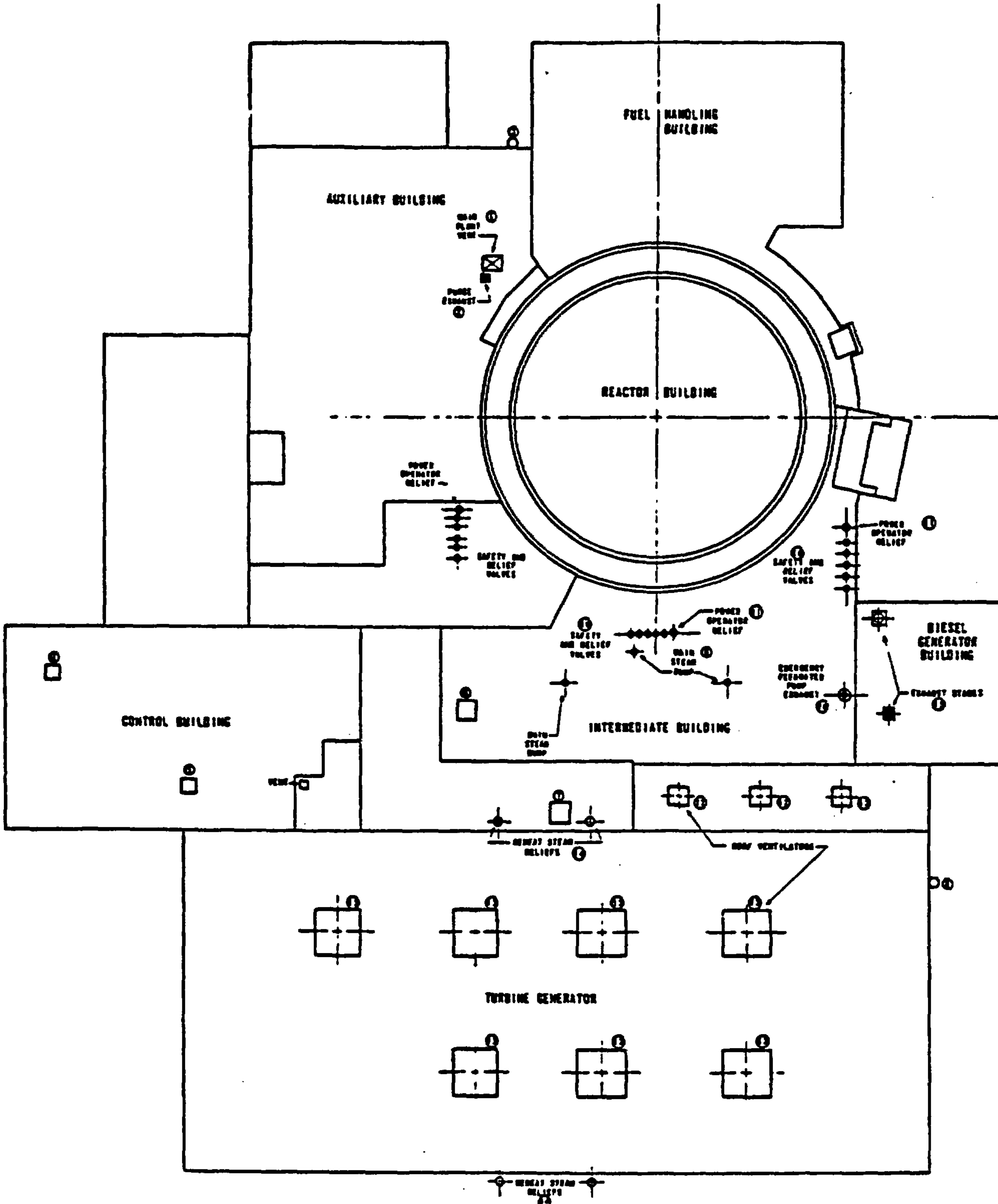
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Figure 5.1-1
Site Area Map



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Figure 5.1-2
Plan of the Virgil C. Summer Nuclear Station

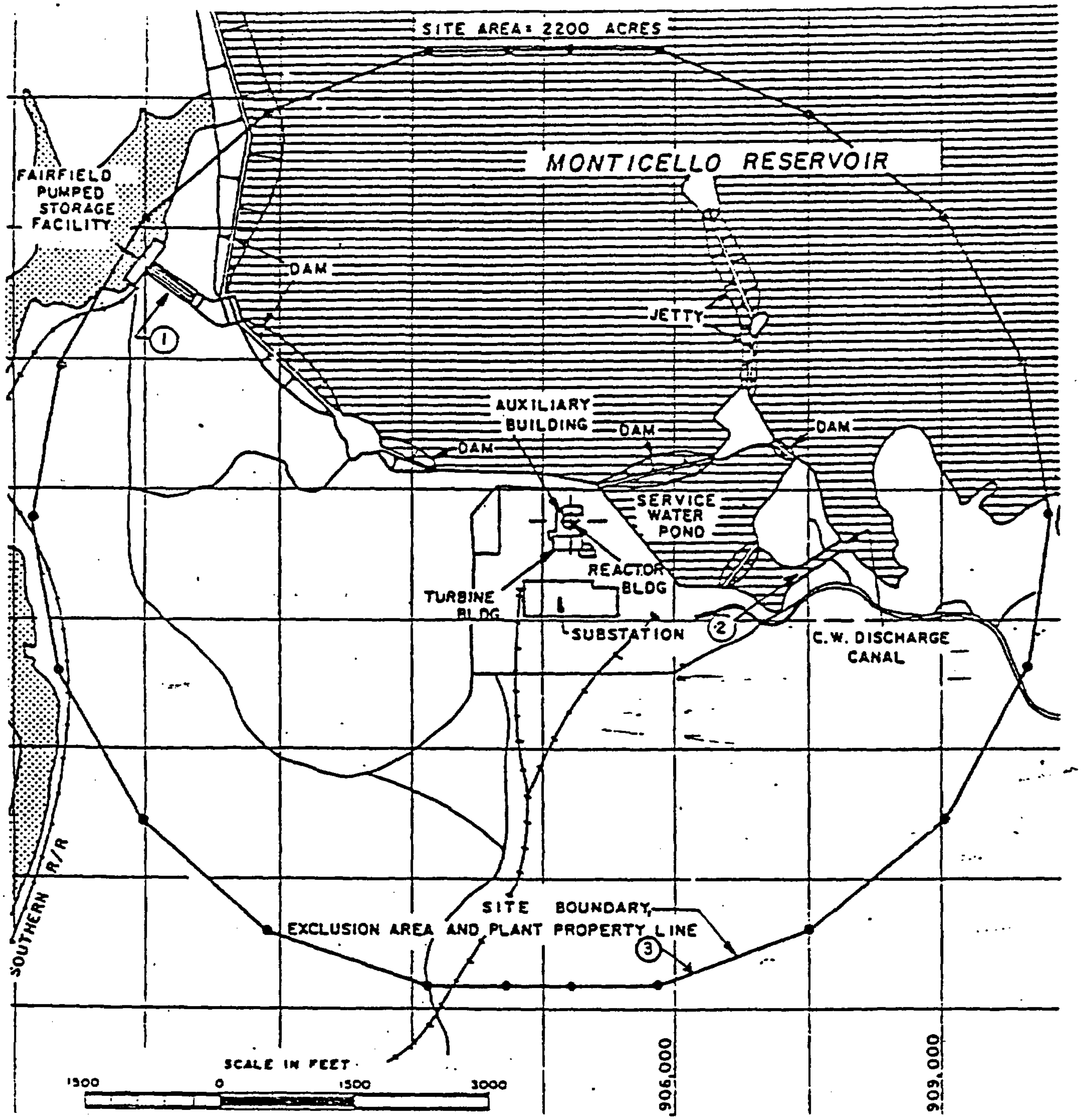


NOTE: See Figure 5.1-4 for site boundary for gaseous effluents

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Potentially Radioactive Gaseous Waste Release Points

FIGURE 5.1-3



LIQUID RELEASES:

1. FAIRFIELD PUMPED STORAGE FACILITY PENSTOCKS
   (A) LIQUID WASTE PROCESSING SYSTEM
   (B) PROCESSED STEAM GENERATOR BLOWDOWN
2. CIRCULATING WATER DISCHARGE CANAL
   (A) UNPROCESSED STEAM GENERATOR BLOWDOWN
   (B) TURBINE BUILDING FLOOR DRAINS

GASEOUS RELEASES:

3. SITE BOUNDARY FOR GASEOUS RELEASES

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Location of Liquid Release Points

FIGURE 5.1-4

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies. Each fuel assembly shall consist of 264 Zircaloy-4, ZIRLO™, or Optimized ZIRLO™ clad fuel rods with an initial composition of uranium dioxide with a maximum nominal enrichment of 4.95 weight percent U-235 as fuel material. Limited substitutions of Zircaloy-4, ZIRLO™, or Optimized ZIRLO™ and/or stainless steel filler rods for fuel rods, if justified by a cycle specific reload analysis using an NRC-approved methodology, may be used. Fuel assembly configurations shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or cycle-specific reload analyses to comply with all fuel safety design bases. Reload fuel shall contain sufficient integral fuel burnable absorbers such that the requirements of Specifications 5.6.1.1a.2 and 5.6.1.2.b are met. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core locations.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9914 ± 100 cubic feet at an indicated T_{avg} of 587.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks consist of 1712 individual storage cells. The cells are grouped into two regions, which are determined based on storage cell spacing as defined below. The spent fuel storage racks are designed, and shall be maintained, with a K_{eff} less than or equal to 0.95 when flooded with unborated water, which includes conservative allowances for uncertainties and biases. This is ensured by maintaining the following for each region:

a. REGION 1 - designated for storage of fresh fuel assemblies and fuel assemblies with a cumulative burnup less than the required cumulative burnup for storage in Region 2.

   1. A nominal 10.867 inch center-to-center distance between fuel assemblies placed in the storage rack.
   2. A maximum nominal initial enrichment of 4.95 weight percent U-235.

b. REGION 2 - designated for storage of discharged fuel assemblies.

   1. A nominal 9.07 inch center-to-center distance between fuel assemblies placed in the storage rack.
   2. A cumulative burnup with the acceptable domain defined by Figure 3.7-1.

5.6.1.2 The new fuel storage racks consist of 60 individual cells, each of which accommodates a single assembly. The new fuel pit storage racks are designed and shall be maintained with a K_{eff} less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 for low density optimum moderation conditions, including conservative allowances for uncertainties and biases. This is ensured by maintaining:

a. A nominal 21 inch center-to-center distance between new fuel assemblies placed in the storage rack.

b. A nominal enrichment of 5.0 weight percent U-235.

DRAINAGE

5.6.2 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 460'3".

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1712 fuel assemblies, with 200 assemblies in Region 1 and 1512 assemblies in Region 2.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/hr and 200 cooldown cycles at < 100°F/hr.	Heatup cycle - T_{avg} from at ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/hr.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate turbine or reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical ESF Electrical System.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 inadvertent auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	50 leak tests.	Pressurized to ≥ 2485 psig.
	5 hydrostatic pressure tests.	Pressurized to ≥ 3107 psig.
	200 large stepload decrease with steam dump.	Load decreases of more than 10% RATED THERMAL POWER occurring in 1 minute or less.
Secondary System	1 steam line break.	Break in a > 6 inch steam line.
	5 hydrostatic pressure tests.	Pressurized to ≥ 1350 psig.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Manager, Nuclear Plant Operations 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 Manager shall be responsible for unit operations. A management directive to this effect, signed by the Vice President, Nuclear Operations, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Offsite and Onsite organizations shall be established for unit operation and corporate management, respectively. The offsite and onsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest 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, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. The organizational charts will be documented in the FSAR and updated in accordance with 10 CFR 50.71(e).

b. The General Manager, Nuclear Plant Operations, shall be responsible for overall unit safe operation and shall have control over onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President, Nuclear Operations, shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out the health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one Licensed Senior Reactor Operator shall be in the Control Room.

c. A health physics technician# shall be on site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. Deleted.

The health physics technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SUMMER UNIT 1

POSITION	**NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION**	
	MODES 1, 2, 3, & 4	**MODES 5 & 6**
SM	1	1
CRF	1	None
RO	2	1
AO	2	1
STA	1	None

SM - Shift Manager with a Senior Reactor Operators License on Unit 1
CRF - Control Room Supervisor with a Senior Reactor Operators License on Unit 1
RD - Individual with a Reactor Operators License on Unit 1
AD - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Manager, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Control Room Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Manager from the Control Room while the unit is in MODE 5 or 6, an individual with a valid RO or SRO license shall be designated to assume the Control Room command function.

6.2.3 NOT USED

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall provide technical support to the Shift Manager in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 NOT USED

The following pages were deleted:
6-4
6-5
6-6
6-7
6-8
6-9

6.5 DELETED

6.6 NOT USED

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President, Nuclear Operations and the NSRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRC and the Vice President, Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.
b. Refueling operations.
c. Surveillance and test activities of safety-related equipment.
d. Security Plan.
e. Emergency Plan.
f. PROCESS CONTROL PROGRAM.
g. OFFSITE DOSE CALCULATION MANUAL.
h. Effluent and environmental monitoring program using the guidance in Regulatory Guide 4.15, Revision 1, February 1979.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed prior to implementation as set forth in 6.5 above.

6.8.3 NOT USED.

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 chemical and volume control, letdown, safety injection, residual heat removal, nuclear sampling, liquid radwaste handling, gas radwaste handling and reactor building spray system. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and
2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

1) Training of personnel,
2) Procedures for monitoring, and
3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective actions for all off-control point chemistry conditions,

6) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Not Used

e. 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 determinations in accordance with the methodology in the ODCM;

2) Limitations on the concentration of radioactive material released in liquid effluents to unrestricted areas conforming to 10 times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2;

e. Radioactive Effluent Controls Program (Continued)

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

4) Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50;

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases or radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual or dose commitment conforming to Appendix I to 10 CFR Part 50;

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

(a) For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin; and

(b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ;

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50;

9) Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate from with half-lives greater than 8 days in gaseous effluents released to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50;

10) Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

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f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measures of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

2) A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of the census; and

3) Participation in an Inter-laboratory Comparison Program to ensure that independent checks on the precision and accuracy of measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

g. Containment Leakage Rate Testing Program

A program shall be established to implement leakage rate testing of the containment system as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995; NEI 94-01, "Industry Guideline for Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, July 2012; ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements"; as modified by approved exceptions that the next Type A test performed after the October 15, 2003 Type A test shall be performed no later than October 15, 2018.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 45.1 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.20 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

g. Containment Leakage Rate Testing Program (Continued)

2) Air lock testing acceptance criteria are:

a. Overall air lock leakage rate is $\leq 0.10\ L_a$ when tested at $\geq P_a$.

b. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 8.0 psig for at least 3 minutes.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

h. Containment Inservice Inspection Program

This program provides controls for monitoring containment vessel structural integrity including routine inspections and tests to identify degradation and corrective actions if degradation is found. The Containment Inservice Inspection Program, inspection frequencies and acceptance criteria shall be in accordance with 10CFR50.55a as modified by approved exemptions. Predicted lift-off forces shall be determined consistent with the recommendations of Regulatory Guide 1.35.1, Revision 3 dated July 1990.

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inservice Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion of necessary repairs, and the extent, nature, and frequency of additional examinations.

In addition, any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with Technical Specification 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

i. 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 shall be made under appropriate administrative control 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 FSAR or bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to insure that the Bases are maintained consistent with the FSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.i.2.b above shall be reviewed and approved 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).

j. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Positions C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

k. Steam Generator Program

A Steam Generator Program shall be established and implemented to ensure that steam generator (SG) tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

1. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

2. Performance criteria for SG tube integrity. Steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   a) Structural integrity performance criterion. All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, HOT STANDBY, and cooldown), all anticipated transients included in the design specifications, and design basis accidents. This includes retaining a safety factor of 3.0 (3deltaP) against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   b) Accident induced leakage performance criterion. The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm per SG.

   c) The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

3. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

4. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of 4.a, 4.b, and 4.c below, the inspection scope, inspection methods and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

a) Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

b) After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in 1), 2), 3), and 4) below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

   1) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

   2) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

   3) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

   4) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

c) If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

5. Provisions for monitoring operational primary-to-secondary leakage.

I. 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 accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989.

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 < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency Filtration System	21,270 SCFM
Reactor Building Cooling Units	60,270 ACFM

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency Filtration System	21,270 SCFM

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 2, 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 (86°F) and the relative humidity specified below.

ESF Ventilation System	Penetration	RH	Face Velocity (fps)
Control Room	<2.5%	70%	0.667

4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below ± 10%.

ESF Ventilation System	Delta P	Flowrate
Control Room	<6 in. W.G.	21,270 SCFM
Reactor Building Cooling Units	<3 in. W.G.	60,270 ACFM

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

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 Filtration System (CREFS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

1. The definition of the CRE and the CRE boundary.

2. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

3. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

4. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, operating at the flow rate required by the Ventilation Filter Testing Program (VFTP), at a Frequency of 36 months on a STAGGERED TEST BASIS such that one train is tested every 18 months. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

5. 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 6.8.4.m.3. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

6. The provisions of 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 6.8.4.m.3 and 6.8.4.m.4, respectively.

n. 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 Vessel (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval.

3) The program shall, as allowed by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements," and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," or meet authorized alternatives pursuant to 10 CFR 50.55a(a)(3).

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 limitations and modifications listed therein.

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 Regional Administrator Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions an 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 REPORT

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 31 of each year. The initial report shall be submitted prior to March 31 of the year following initial criticality.

6.9.1.5 Not used.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 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 an 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.

6.9.1.7 Not used.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 Annual radioactive effluent release report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

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6.9.1.9 Not used.

6.9.1.10 Not used.

CORE OPERATING LIMITS REPORT

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Moderator Temperature Coefficient BOL and EOL Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

b. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

c. Control Rod Insertion Limits for Specification 3/4.1.3.6,

d. Axial Flux Difference Limits, target band, and APL^{ND} for Specification 3/4.2.1,

e. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(z), W(z), APL^{ND}, $W(z)_{BL}$, and $F_Q(z)$ manufacturing/measurement uncertainties for Specification 3/4.2.2,

f. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, Power Factor Multiplier, $PF_{\Delta H}$, and $F_{\Delta H}^{N}$ measurement uncertainties limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Rod Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (F_Q Methodology for W(z) surveillance requirements).)

CORE OPERATING LIMITS REPORT (Continued)

c. WCAP-12945-P-A, Volume 1 (Revision 2) through Volumes 2 through 5 (Revision 1) "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

Liparulo, N. (W) to NRC Document Control Desk, NSD-NRC-96-4746, "Re-Analysis Work Plans Using Final Best Estimate Methodology" dated 6/13/1996.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," August 1994, (W Proprietary).

WCAP-12472-P-A, Addendum 1-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM," January 2000, (W Proprietary)

(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (Westinghouse Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

f. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995 (W Proprietary). WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.12 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.k. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Office of Inspection and Enforcement Regional Office 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 Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREAS

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr* 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). Health Physics personnel or individuals escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned duties, provided they otherwise comply with approved radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device 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 RWP.

* Measurement made at 30 cm (12 in.) from the radiation source or from any surface penetrated by the radiation.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr* but less than 500 rads/hr** shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the duty Shift Manager 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. The maximum allowable stay time for individuals in that area shall be established prior to entry. In lieu of the stay time specification of the RWP, direct or remote continuous surveillance (such as closed circuit TV cameras) shall be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

For individual areas accessible to personnel with radiation levels greater than 1000 mrem/hr* but less than 500 rads/hr** that are located within larger areas (such as PWR containment) where no enclosure can be reasonably constructed around the individual areas, then those areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained for the duration of the Unit Operating License. This documentation shall contain:

    1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s); and

    2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the PSRC and approval of the General Manager, Nuclear Plant Operations.

* Measurement made at 30 cm (12 in.) from the radiation source or from any surface penetrated by the radiation.

** Measurement made at 1 meter from the radiation source or from any surface penetrated by the radiation.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained for the duration of the Unit Operating License. This documentation shall contain:

   1) Sufficient information to support the change together with appropriate analyses or evaluations justifying the change(s); and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent dose or setpoint calculations.

b. Shall become effective after review and acceptance by the PSRC and the approval of the General Manager, Nuclear Plant Operations.

c. Shall be submitted to the Commission in the form of a complete legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

APPENDIX B

TO RENEWED FACILITY LICENSE NO. NPF-12

FOR

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

ENVIRONMENTAL PROTECTION PLAN

VIRGIL C. SUMMER NUCLEAR STATION

UNIT NO. 1

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

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1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during construction and operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement (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 FES which relate to water quality matters are regulated by means of the licensees' NPDES permit.

2.0 Environmental Protection Issues

In the FES-OL dated May 1981, the staff has considered the environmental impacts associated with the operation of the Virgil C. Summer Nuclear Station, Unit No. 1. No environmental issues were identified which require license conditions to resolve environmental concerns and to assure adequate protection of the environment.

Aquatic monitoring is addressed by the effluent limitations, monitoring requirements and demonstration studies contained in the effective NPDES permit issued by the South Carolina Department of Health and Environmental Control. The NRC will rely on that agency for regulations of matters involving water quality and aquatic biota.

3.0 Consistency Requirements

3.1 Facility Design and Operation

The licensees may make changes in facility design or operation or perform tests or experiments affecting the environment provided that such changes, tests or experiments do not involve an unreviewed environmental question. Changes in facility design or operation or performance of tests or experiments which do not affect the environment are not subject to this requirement. Activities governed by Section 3.3 are not subject to the requirements of this section.

Before engaging in unauthorized construction or operational activities which may affect the environment, the licensees shall prepare and record an environmental evaluation of such activity.[2] When the evaluation indicates that such activity involves an unreviewed environmental question, the licensees 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 that may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the 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 paragraph, which may have a significant adverse environmental impact.

The licensees 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

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[2]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.

evaluation which provides bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question.

3.2 Reporting Related to the NPDES Permits and State Certifications

Violations of the NPDES permit or State certification (pursuant to Section 401 of the Clean Water Act) shall be reported to the NRC by submittal of copies of the reports required by the NPDES permit or certification. South Carolina Electric & Gas Company shall also provide the NRC with a copy of the results of the following studies at the same time they are submitted to the permitting agency:

(1) Thermal Effects Study Plan

(2) Section 316(b) Demonstration Study

Changes and additions to the NPDES permit or the State 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.

The NRC shall be notified of changes to the effective NPDES permit proposed by the licensees by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. South Carolina Electric & Gas Company shall provide the NRC with a copy of the application for renewal of the NPDES permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in facility design or operation and performance of tests or experiments which are required to achieve compliance with 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 that indicates or could result in significant environmental impact causally related to facility operation shall be recorded and promptly reported to the NRC within 24 hours followed by a written report within 30 days. No routine monitoring programs are required to implement this condition.

The written report shall (a) describe, analyze, and evaluate the event, including the extent and magnitude of the impact and facility operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (d) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided with a copy of such reports at the same time they are submitted to the other agencies.

The following are examples of unusual or important events: excessive bird impaction events; onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; unusual fish kills; increase in nuisance organisms or conditions; and unanticipated or emergency discharge of waste water or chemical substance.

APPENDIX C

ADDITIONAL CONDITIONS
RENEWED OPERATING LICENSE NO. NPF-12

South Carolina Electric & Gas Company (the term licensee in Appendix C refers to South Carolina Electric & Gas Company) shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
137	The licensee is authorized to relocate certain Technical Specification requirements to license-controlled documents. Implementation of this amendment shall include the relocation of those technical specification requirements to the appropriate documents, as described in the licensee's application dated November 14, 1995, as supplemented by letters dated July 11, 1996, and July 24, 1997, and evaluated in the staff's Safety Evaluation attached to this amendment.	The amendment shall be implemented within 180 days from August 13, 1997.
180	Upon implementation of Amendment No. 180 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by Surveillance Requirement SR 4.7.6.e. in accordance with TS 6.8.4.m.3.(i), the assessment of CRE habitability as required by Specification 6.8.4.m.3.(ii), and the measurement of CRE pressure as required by Specification 6.8.4.m.4, shall be considered met. Following implementation: (a) The first performance of SR 4.7.6, in accordance with Specification 6.8.4.m.3.(i), shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from March 25, 2005, the date of the most recent successful tracer gas test, as stated in the November 18, 2005 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. (b) The first performance of the periodic assessment of CRE habitability, specification 6.8.4.m.3.(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from March 25, 2005, the date of the most recent successful tracer gas test, as stated in the November 18, 2005 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years. The first performance of the periodic measurement of CRE pressure, specification 6.8.4.m.4, shall be within 18 months, plus the 138 days allowance of SR 4.0.2, as measured from September 28, 2006, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.	As stated in the Additional Condition