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

ENTERGY ARKANSAS, LLC

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

1. The Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License NPF-6 issued on September 1, 1978 has now found that:

   A. The application to renew License NPF-6 filed by Entergy Arkansas, LLC and Entergy Operations, Inc. (EOI), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of Arkansas Nuclear One, Unit 2 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-89 and the application, as amended, the provisions of the Act and the regulations of the Commission.

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

   D. The facility requires exemptions from certain requirements of (1) Sections 50.55a(g)(2) and 50.55a(g)(4) of 10 CFR Part 50, (2) Appendices G and H to 10 CFR Part 50, and (3) Appendix J to 10 CFR Part 50 for a period of three years. These exemptions are described in the Office of Nuclear Reactor Regulation's safety evaluations supporting the granting of these exemptions which are enclosed in the letter transmitting this license amendment. 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 exemptions are, therefore, hereby granted. With the granting of these exemptions, the facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

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

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

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

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

I. After weighing the environmental, economic, technical and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Renewed Facility Operating License No. NPF-6 subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 (formerly Appendix D to 10 CFR Part 50) of the Commission's regulations and all applicable requirements have been satisfied; and

J. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70, including 10 CFR Sections 30.33, 40.32, 70.23 and 70.31.

2. Facility Operating License No. NPF-6, issued September 1, 1978, is superceded by Renewed Facility Operating License No. NPF-6, which is hereby issued to Entergy Arkansas, LLC and Entergy Operations, Inc. to read as follows:

A. This renewed license applies to Arkansas Nuclear One, Unit 2, a pressurized water reactor and associated equipment (the facility) owned by Entergy Arkansas, LLC. The facility is located in Pope County, Arkansas and is described in the Final Safety Analysis Report as supplemented and amended (Amendments 20 through 47) and the Environmental Report as supplemented and amended (Amendments 1 through 7).

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

(1) Entergy Arkansas, LLC pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, but not operate, the facility at the designated location in Pope County, Arkansas in accordance with the procedures and limitations set forth in this renewed license.

(2) EOI, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in Pope County, Arkansas in accordance with the procedures and limitations set forth in this renewed license;

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

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

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

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

C. This renewed license shall be deemed to contain and is subject to conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 3026 megawatts thermal. Prior to attaining this power level EOI shall comply with the conditions in Paragraph 2.C.(3).

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 335, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

Exemptive 2nd paragraph of 2.C.2 deleted per Amendment 20, 3/3/81.

(3) <u>Additional Conditions</u>

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission.

2.C.(3)(a) Deleted per Amendment 24, 6/19/81.

(b) Fire Protection

Entergy Operations, Inc. 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 license amendment requests dated December 17, 2012, and October 27, 2016, and supplements dated November 7, 2013, December 4, 2013, January 6, 2014, May 22, 2014, June 30, 2014, August 7, 2014, September 24, 2014, December 9, 2014, December 2, 2016, and February 21, 2017, and as approved in the SEs dated February 18, 2015, and May 12, 2017. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at ANO-2. 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 $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

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

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

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

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

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

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

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC SE dated February 18, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the Entergy Operations, Inc. fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.
2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications," Attachment 2, of Entergy Operations, Inc. letter 2CAN101601, dated October 27, 2016, prior to startup from the second refueling outage following issuance of the Safety Evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of the modifications.
3. The licensee shall complete the implementation items as listed in Table S-2, "Implementation Items," Attachment, of Entergy Operations, Inc. letter 2CAN091402, dated September 24, 2014, within six months after issuance of the Safety Evaluation.

(c) Less Than Four Reactor Coolant Pump Operation

EOI shall not operate the reactor in operational Modes 1 and 2 with fewer than four reactor coolant pumps in operation, except as allowed by Special Test Exception 3.10.3 of the facility Technical Specifications.

(d) Surveillance Frequency Control Program

The licensee shall implement the items listed in Table 2 of the enclosure to Entergy letter 2CAN111801, dated November 16, 2018, prior to implementation of the Surveillance Frequency Control Program.

2.C.(3)(e) Deleted per Amendment 300, 2/18/15.

2.C.(3)(f) Deleted per Amendment 24, 6/19/81.

2.C.(3)(g) Deleted per Amendment 93, 4/25/89.

2.C.(3)(h) Deleted per Amendment 29, (3/4/82) and its correction letter, (3/15/82).

(i) Containment Radiation Monitor

AP&L shall, prior to July 31, 1980 submit for Commission review and approval documentation which establishes the adequacy of the qualifications of the containment radiation monitors located inside the containment and shall complete the installation and testing of these instruments to demonstrate that they meet the operability requirements of Technical Specification No. 3.3.3.6.

2.C.(3)(j) Deleted per Amendment 7, 12/1/78.

2.C.(3)(k) Deleted per Amendment 12, 6/12/79 and Amendment 31, 5/12/82.

2.C.(3)(l) Deleted per Amendment 24, 6/19/81.

2.C.(3)(m) Deleted per Amendment 12, 6/12/79.

2.C.(3)(n) Deleted per Amendment 7, 12/1/78.

2.C.(3)(o) Deleted per Amendment 7, 12/1/78.

2.C.(3)(p) Deleted per Amendment 255, 9/28/04.

2.C.(4) (Number has never been used.)

2.C.(5) Deleted per Amendment 255, 9/28/04.

2.C.(6) Deleted per Amendment 255, 9/28/04.

2.C.(7) Deleted per Amendment 78, 7/22/86.

(8) Antitrust Conditions

EOI shall not market or broker power or energy from Arkansas Nuclear One, Unit 2. Entergy Arkansas, LLC is responsible and accountable for the actions of its agents to the extent said agent's actions affect the marketing or brokering of power or energy from ANO, Unit 2.

(9) Rod Average Fuel Burnup

Entergy Operations is authorized to operate the facility with an individual rod average fuel burnup (burnup averaged over the length of a fuel rod) not to exceed 60 megawatt-days/kilogram of uranium.

(10) Mitigation Strategies

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

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

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

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

(11) Upon implementation of Amendment 288 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 4.7.6.1.2.d, in accordance with Specifications 6.5.12.c.(i), 6.5.12.c.(ii), and 6.5.12.d, shall be considered met. Following implementation:

(i) The first performance of SR 4.7.6.1.2.d, in accordance with Specification 6.5.12.c.(i), shall be within 15 months of the approval of TSTF-448. SR 4.0.2 will not be applicable to this first performance.

(ii) The first performance of the periodic assessment of CRE habitability, Specification 6.5.12.c.(ii), shall be within 15 months of the approval of TSTF-448. SR 4.0.2 will not be applicable to this first performance.

(iii) The first performance of the periodic measurement of CRE pressure, Specification 6.5.12.d, shall be within 15 months of the approval of TSTF-448. SR 4.0.2 will not be applicable to this first performance.

D. Physical Protection

EOI 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 contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Physical Security, Safeguards Contingency and Training & Qualification Plan," as submitted on May 4, 2006.

EOI 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 EOI CSP was approved by License Amendment No. 294 as supplemented by changes approved by License Amendment Nos. 295, 298, and 303.

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

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, EOI will prepare and record an environmental evaluation for such activity. When the evaluation indicates that such activity 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 (NUREG-0254) or any addendum thereto, and other NRC environmental impact assessments, EOI shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation.

F. Updated Final Safety Analysis Report Supplement

The Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, ANO-2 may make changes to the programs and activities described in the supplement without prior Commission approval, provided that ANO-2 evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements of that section.

The ANO-2 Final Safety 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. ANO-2 shall complete these activities no later than July 17, 2018, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

G. Reactor Vessel Material Surveillance Capsules

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

H. 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors

Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except wind-generated missiles and seismic; the tornado safe shutdown equipment list for wind-generated missiles; and the alternative seismic approach as described in the Entergy submittal letter dated May 26, 2021, and all its subsequent associated supplements, as specified in License Amendment No. 331 dated July 19, 2022.

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

4. This renewed license is effective as of the date of issuance and shall expire at midnight, July 17, 2038.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by J. E. Dyer

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

Attachments:

1. Appendix A - Technical Specifications
2. Preoperational Tests, Startup Tests and other items which must be completed by the indicated Operational Mode

Date of Issuance: June 30, 2005

NUREG 0336

LICENSE AUTHORITY FILE COPY

Transmitted with Original License dtd. 7-18-78

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ARKANSAS NUCLEAR ONE

UNIT 2

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

LICENSE NO. NPF-6

SECTION 1.0
DEFINITIONS

DEFINED TERMS

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

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3026 MWt.

OPERATIONAL MODE – MODE

1.4 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.

ACTION

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

OPERABLE – OPERABILITY

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

REPORTABLE OCCURRENCE

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

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

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

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each airlock is OPERABLE pursuant to Specification 3.6.1.3,

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

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

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels – The injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY.

b. Bistable channels – The injection of a simulated signal into the sensor to verify OPERABILITY of all devices in the channel required for channel OPERABILITY.

c. Digital computer channels – The exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY of all devices in the channel required for channel OPERABILITY.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control element assemblies are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

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

b. Leakage into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of leakage detection systems, or

c. Reactor coolant system leakage through a steam generator to the secondary system (primary to secondary leakage).

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or controlled leakage.

PRESSURE BOUNDARY LEAKAGE

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

AZIMUTHAL POWER TILT – T_q

1.17 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

DOSE EQUIVALENT I-131

1.18 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The CEDE dose conversion factors used to determine the DOSE EQUIVALENT I-131 shall be performed using Table 2.1 of EPA Federal Guidance Report No. 11, 1988, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion.”

DOSE EQUIVALENT XE-133

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

1.20 Deleted

FREQUENCY NOTATION

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

AXIAL SHAPE INDEX

1.22 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

REACTOR TRIP SYSTEM RESPONSE TIME

1.23 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.24 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 methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.

PHYSICS TESTS

1.25 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.

SOFTWARE

1.26 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation and procedures.

PLANAR RADIAL PEAKING FACTOR F_{xy}

1.27 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

LIQUID RADWASTE TREATMENT SYSTEM

1.28 A LIQUID RADWASTE TREATMENT SYSTEM is a system designed and installed to reduce radioactive liquid effluents from the unit. This is accomplished by providing for holdup, filtration, and/or demineralization of radioactive liquid effluents prior to their release to the environment.

MEMBER(S) OF THE PUBLIC

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

PURGE – PURGING

1.30 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to reduce airborne radioactive concentrations in such a manner that replacement air or gas is required to purify the confinement.

EXCLUSION AREA

1.31 The EXCLUSION AREA is that area surrounding ANO within a minimum radius of .65 miles of the reactor buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

UNRESTRICTED AREA

1.32 An UNRESTRICTED AREA shall be any area at or beyond the exclusion area boundary.

CORE OPERATING LIMITS REPORT

1.33 The CORE OPERATING LIMITS REPORT is the ANO-2 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification 6.6.5. Plant operation within these operating limits is addressed in individual specifications.

INSERVICE TESTING PROGRAM

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

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	≥ 300 °F
2. STARTUP	≥ 0.99	$\leq 5\%$	≥ 300 °F
3. HOT STANDBY	< 0.99	0	≥ 300 °F
4. HOT SHUTDOWN	< 0.99	0	300 °F > T_{avg} > 200 °F
5. COLD SHUTDOWN	< 0.99	0	≤ 200 °F
6. REFUELING**	≤ 0.95	0	≤ 140 °F

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S/U	Prior to each reactor startup
N.A.	Not applicable
SFCP	In accordance with the Surveillance Frequency Control Program

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained ≥ 1.23.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor core has decreased to less than 1.23, be in HOT STANDBY within 1 hour.

PEAK FUEL CENTERLINE TEMPERATURE

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-275-P, Revision 1-P-A and CENPD-382-P-A).

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak fuel centerline temperature has equaled or exceeded 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-275-P, Revision 1-P-A and CENPD-382-P-A), be in HOT STANDBY within 1 hour.

SAFETY LIMITS

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

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

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

ACTION:

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

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Correction letter of 3-5-86

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ 110% of RATED THERMAL POWER	≤ 110.712% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.75%	≤ 0.819%
4. Pressurizer Pressure - High	≤ 2362 psia	≤ 2370.887 psia
5. Pressurizer Pressure - Low	≥ 1650 psia (2)	≥ 1618.9 psia (2)
6. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
7. Steam Generator Pressure - Low	≥ 751 psia (3)	≥ 738.6 psia (3)
8. Steam Generator Level - Low	≥ 22.2% (4)	≥ 21.5% (4)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High	≤ 21.0 kw/ft (5)	≤ 21.0 kw/ft (5)
10. DNBR - Low	≥ 1.25 (5)	≥ 1.25 (5)

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% power; bypass shall be automatically removed before decreasing below 10^{-4}% power.

(2) Value may be decreased manually, to a minimum value of 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed before pressurizer pressure exceeds 500 psia.

(3) Value may be decreased manually during a planned reduction in steam generator pressure provided the margin between the steam generator pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower narrow range level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-2}% power; bypass shall be automatically removed before exceeding 10^{-2}% power.

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation (LCO) and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification, except as provided in LCO 3.0.2, LCO 3.0.8, and LCO 3.0.9.

3.0.2 Adherence to the requirements of the LCO and/or associated ACTION within the specified time interval shall constitute compliance with the specification, except as provided in LCO 3.0.5 and LCO 3.0.6. In the event the LCO is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification 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 at least HOT STANDBY within 6 hours, in at least HOT SHUTDOWN within the next 6 hours, and in at least COLD SHUTDOWN within the following 24 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial discovery or until the reactor is placed in a MODE in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specification.

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

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

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

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

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

3/4.0 APPLICABILITY (continued)

LIMITING CONDITION FOR OPERATION

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

3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the ACTIONs associated with this supported system are not required to be entered. Only the support system LCO ACTIONs are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 6.5.19, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate ACTIONs of the LCO in which the loss of safety function exists are required to be entered.

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

3.0.7 To be used later.

3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

3.0.9 When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

LIMITING CONDITION FOR OPERATION

3.0.9 (continued)

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the Surveillance. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified interval shall be failure to meet the LCO except as provided in 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

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

4.0.3 If it is discovered that a Surveillance was not performed within its specified interval, 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 interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. 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 a MODE or other specified condition in the Applicability shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided in SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN – T_{avg} > 200 °F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution 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 that specified in the CORE OPERATING LIMITS REPORT:

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

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

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA 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 (e) below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

---

* See Special Test Exception 3.10.1.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% Δk/k in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution 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 that specified in the CORE OPERATING LIMITS REPORT:

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

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

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

Next page is 3/4 1-5

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT. The maximum upper design limit shall be:

a. Less positive than $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ whenever THERMAL POWER is ≤ 70% of RATED THERMAL POWER, and

b. Less positive than 0.0 $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits. (Note 1)

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

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading. (Note 1)

b. At greater than 5% of RATED THERMAL POWER, within 7 effective full power days (EFPD) of reaching 40 EFPD core burnup.

c. At greater than 5% of RATED THERMAL POWER, within 7 EFPD of reaching two-thirds of expected core burnup. (Note 2)

---

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

See Special Test Exception 3.10.2.

Note 1: For fuel cycles that meet the applicability requirements given in WCAP-16011-P-A, the verification prior to entering MODE 1 may be made using the predicted MTC as adjusted for the measured boron concentration.

Note 2: The MTC determination of surveillance 4.1.1.4.2.c is not required if the results of the tests required in surveillances 4.1.1.4.2.a and 4.1.1.4.2.b are within a tolerance of $\pm 0.16 \times 10^{-4}$ $\Delta k/k/°F$ from the corresponding design values. For cycles that meet the applicability requirements given in WCAP-16011-P-A, the MTC determination of surveillance 4.1.1.4.2.c is not required if the result of the test required in surveillance 4.1.1.4.2.b is within a tolerance of $\pm 0.16 \times 10^{-4}$ $\Delta k/k/°F$ from the corresponding design value.

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be ≥ 540 °F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 540 °F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be: ≥ 540 °F in accordance with the Surveillance Frequency Control Program.

With K_{eff} ≥ 1.0.

* See Special Test Exception 3.10.5.

3/4.1.3 CONTROL ELEMENT ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All CEAs shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more CEA(s) 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 at least HOT STANDBY within the next 6 hours.

b. With one CEA trippable but inoperable due to causes other than addressed by ACTION (a) above, but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specifications 3.1.3.5 and 3.1.3.6.

c. With more than one CEA trippable but inoperable due to causes other than addressed by ACTION (a) above, but within the above specified alignment requirements, restore the inoperable CEA(s) to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours.

d. With one CEA trippable but misaligned from any other CEA in its group by more than 7 inches, operation in MODES 1 and 2 may continue, provided that, for inward deviations, core power is reduced in accordance with the limits specified in the CORE OPERATING LIMITS REPORT and, for all deviations, within 2 hours either:

   1. Restore the misaligned CEA to within its above specified alignment requirements, or

   2. Verify the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. Operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.5 and 3.1.3.6 provided:

      a) Within two hours following the misalignment the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT, and

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

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

*See Special Test Exceptions 3.10.2 and 3.10.4.

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ACTION: (Continued)

e. With more than one CEA misaligned from any other CEA in its group by more than 7 inches (indicated position), be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group in accordance with the Surveillance Frequency Control Program.

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction in accordance with the Surveillance Frequency Control Program.

Note 1 - Movement of CEA 4 is not required for the remainder of Cycle 26. If an outage of sufficient duration occurs prior to the end of Cycle 26, maintenance activities will be performed to restore the CEA.

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POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any CEA having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore at least one of the inoperable position indicator channels to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA with the inoperable position indicators at its fully withdrawn or fully inserted position while maintaining the requirements of Specifications 3.1.3.1, 3.1.3.5, and 3.1.3.6. Operation may then continue provided the CEA with the inoperable position indicators is maintained fully withdrawn or fully inserted, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and the CEA position is verified at least once per 12 hours thereafter by its "Full Out" or "Full In" limit.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5 inches of each other in accordance with the Surveillance Frequency Control Program.

POSITION INDICATOR CHANNELS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each CEA not fully inserted.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

* With the reactor trip breakers in the closed position.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual CEA drop time, from a fully withdrawn position, shall be ≤ 3.9 seconds and the arithmetic average of the CEA drop times of all CEAs, from a fully withdrawn position, shall be ≤ 3.4 seconds from when the electrical power is interrupted to the CEA drive mechanisms until the CEAs reach their 90 percent insertion positions with:

a. $T_{avg} \geq 525$ °F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the CEA drop times determined to exceed either of the above limits, restore the CEA drop times to within the above limits prior to proceeding to MODE 1 or 2.

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. In accordance with the Surveillance Frequency Control Program.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to the Full Out position.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than the Full Out position, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Withdraw the CEA to the Full Out position, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to the Full Out position:

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

b. In accordance with the Surveillance Frequency Control Program.

---

* See Special Test Exception 3.10.2.

With $K_{eff} \geq 1.0$.

REGULATING AND GROUP P CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups and Group P CEAs shall be maintained within the following limits:

a. One or more CEACs OPERABLE:

1. The regulating CEA groups and Group P CEAs shall be limited to the withdrawal sequence and to the insertion limits specified in the CORE OPERATING LIMITS REPORT. CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limit is restricted to:

a) ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

b) ≤ 14 Effective Full Power Days per calendar year.

2. CEA insertion between the Short Term Steady State Insertion Limit and the Transient Insertion Limit shall be restricted to ≤ 4 hours per 24 hour interval.

b. Both CEACs inoperable:

Regulating CEA Group 6 may be inserted no further than 127.5 inches withdrawn which is the Transient Insertion Limit when both CEACs are inoperable. All other CEAs must be maintained fully withdrawn.

APPLICABILITY: MODES 1* and 2*#

ACTION:

a. With the regulating CEA groups or Group P CEAs inserted beyond the Transient Insertion Limit, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours of exceeding the Transient Insertion Limit either:

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

2. Reduce THERMAL POWER as follows:

a) One or more CEACs OPERABLE:

1) Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position specified in the CORE OPERATING LIMITS REPORT, or

2) Be in at least HOT STANDBY within 8 hours of exceeding the Transient Insertion Limit.

* See Special Test Exceptions 3.10.2 and 3.10.4

With $K_{eff} \geq 1.0$.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b) Both CEACs inoperable:

Be in at least HOT STANDBY within 8 hours of exceeding the Transient Insertion Limit.

b. With the regulating CEA groups or Group P CEAs inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:

1. Restore the regulating groups or Group P CEAs to within the Long Term Steady State Insertion Limit within two hours, or

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

c. With the regulating CEA groups or Group P CEAs inserted between the Short Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 4 hours per 24 hour interval, operation may proceed provided any subsequent increase in thermal power is restricted to ≤ 5% of rated thermal power per hour.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group and Group P CEAs shall be determined to be within the Transient Insertion Limits in accordance with the Surveillance Frequency Control Program except during time intervals when the PDIL Alarm is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups or Group P CEAs are inserted beyond the Long Term Steady State Insertion Limit or the Short Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined in accordance with the Surveillance Frequency Control Program.

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has been removed

per

Amendment No. 169

dated

10-12-95

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit shall be maintained by either:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

b. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC Channel (when COLSS is out of service).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With COLSS in service and the linear heat rate limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limit and either:

   1. Restore the linear heat rate to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the linear heat rate limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the linear heat rate to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying in accordance with the Surveillance Frequency Control Program that the linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 In accordance with the Surveillance Frequency Control Program, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on linear heat rate.

POWER DISTRIBUTION LIMITS

RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to PLANAR RADIAL PEAKING FACTOR by a factor equivalent to $\geq F^m_{xy} / F^c_{xy}$ and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}); or

c. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}), obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

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

* See Special Test Exception 3.10.2.

AZIMUTHAL POWER - T_q

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but within the limit specified in the CORE OPERATING LIMITS REPORT, within two hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed the limit specified in the CORE OPERATING LIMITS REPORT:

1. Due to misalignment of a CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

2. Verify that the AZIMUTHAL POWER TILT 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 Linear Power Level - High trip setpoints 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 AZIMUTHAL POWER TILT 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.

Correction letter of 10-24-95

SURVEILLANCE REQUIREMENTS

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt in accordance with the Surveillance Frequency Control Program when the COLSS is inoperable.

c. Verifying in accordance with the Surveillance Frequency Control Program, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors in accordance with the Surveillance Frequency Control Program to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

POWER DISTRIBUTION LIMITS

DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR limit shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and at least one CEAC is operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the value specified in the CORE OPERATING LIMITS REPORT (when COLSS is in service and neither CEAC is operable); or

c. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service and at least one CEAC is operable); or

d. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With COLSS in service and the DNBR limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

   1. Restore the DNBR to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the DNBR limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the DNBR to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying in accordance with the Surveillance Frequency Control Program that the DNBR, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.4.3 In accordance with the Surveillance Frequency Control Program, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

POWER DISTRIBUTION LIMITS

RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 120.4×10^6 lbm/hr.

APPLICABILITY: MODE 1

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The Reactor Coolant Cold Leg Temperature (T_c) shall be maintained between 542 °F and 554.7 °F.

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the Reactor Coolant Cold Leg Temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The Reactor Coolant Cold Leg Temperature shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits in accordance with the Surveillance Frequency Control Program using the COLSS or any OPERABLE Core Protection Calculator channel.

* See Special Test Exception 3.10.2.

PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

APPLICABILITY: MODE 1.

ACTION:

With the average pressurizer pressure exceeding its limits, restore the pressure 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

4.2.8 The average pressurizer pressure shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

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

4.3.1.1.4 The Core Protection Calculator System shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.1.5 The affected Core Protection Calculator Channel shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a valid CPC Cabinet High Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2 2 sets of 2	1 set of 2 1 set of 2	2 sets of 2 2 sets of 2	1,2 3*,4*,5*	5 8
2. Linear Power Level – High	4	2	3	1,2	2,3
3. Logarithmic Power Level – High					
a. Startup	4	2(a)(d)	3	2,3*,4*,5*	2,3
b. Shutdown	4	0	2	3*,4*,5*	4
4. Pressurizer Pressure – High	4	2	3	1,2	2,3
5. Pressurizer Pressure – Low	4	2(b)	3	1,2	2,3
6. Containment Pressure – High	4	2	3	1,2	2,3
7. Steam Generator Pressure – Low	4/SG	2/SG	3/SG	1,2	2,3
8. Steam Generator Level – Low	4/SG	2/SG	3/SG	1,2	2,3
9. Local Power Density – High	4	2(c)(d)	3	1,2	2,3

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. DNBR - Low	4	2(c)(d)	3	1, 2	2,3
11. Reactor Protection System Logic					
A. Matrix Logic	6	1	3	1,2	1
	6	1	3	3*,4*,5*	8
B. Initiation Logic	4	2	4	1,2	5
	4	2	4	3*,4*,5*	8
12. Reactor Trip Breakers	4(f)	2	4	1,2	5
	4(f)	2	4	3*,4*,5*	8
13. Core Protection Calculators	4	2(c)(d)	3	1, 2	2,3,7
14. CEA Calculators	2	1	2(e)	1, 2	6,7

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

(a) Trip may be manually bypassed above 10^{-4}% power; bypass shall be automatically removed before decreasing below 10^{-4}% power.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed before pressurizer pressure exceeds 500 psia.

(c) Trip may be manually bypassed below 10^{-2}% power; bypass shall be automatically removed before exceeding 10^{-2}% power. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 1% power; bypass shall be automatically removed before exceeding 1% power.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

ACTION STATEMENTS

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 2 – With the number of channels OPERABLE one less than the Total Number of Channels, operation in the applicable MODES may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed for greater than 48 hours, the desirability of maintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM). The channel shall be returned to OPERABLE status prior to startup following the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	Functional Unit Bypassed
1. Linear Power (Subchannel or Linear)	Linear Power Level – High Local Power Density – High DNBR – Low Log Power Level – High*
2. Pressurizer Pressure – NR	Pressurizer Pressure – High Local Power Density – High DNBR – Low
3. Containment Pressure – NR	Containment Pressure – High (RPS) Containment Pressure – High (ESFAS) Containment Pressure – High-High (ESFAS)
4. Steam Generator 1 Pressure	Steam Generator 1 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
5. Steam Generator 2 Pressure	Steam Generator 2 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
6. Steam Generator 1 Level	Steam Generator 1 Level – Low Steam Generator 1 ΔP (EFAS 1)
7. Steam Generator 2 Level	Steam Generator 2 Level – Low Steam Generator 2 ΔP (EFAS 2)
8. Core Protection Calculator	Local Power Density – High DNBR – Low

* Only for failure common to both linear power and log power.

ACTION STATEMENTS

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, operation in the applicable MODES may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low Log Power Level - High**
2. Pressurizer Pressure - NR	Pressurizer Pressure - High Local Power Density -High DNBR - Low
3. Containment Pressure - NR	Containment Pressure - High (RPS) Containment Pressure - High (ESFAS) Containment Pressure - High-High (ESFAS)
4. Steam Generator 1 Pressure	Steam Generator 1 Pressure - Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
5. Steam Generator 2 Pressure	Steam Generator 2 Pressure - Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
6. Steam Generator 1 Level	Steam Generator 1 Level - Low Steam Generator 1 ΔP (EFAS 1)
7. Steam Generator 2 Level	Steam Generator 2 Level - Low Steam Generator 2 ΔP (EFAS 2)
8. Core Protection Calculator	Local Power Density - High DNBR - Low

Operation in the applicable MODES may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent operation in the applicable MODES may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

** Only for failure or activities common to both linear power and log power.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 4 – With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 5 – With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, place the reactor trip breakers of the inoperable channel in the tripped condition within 1 hour or be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

ACTION 6 –

a. With one CEAC inoperable, operation may continue for up to 7 days or in accordance with the Risk Informed Completion Time Program provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group. After 7 days or after expiration of the Risk Informed Completion Time, whichever is longer, operation may continue provided that ACTION 6.b is met.

b. With both CEACs inoperable, operation may continue provided that:

   1. Within 1 hour the margin required by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

   2. Within 4 hours:

      a) All CEA groups are withdrawn within the limits of Specifications 3.1.3.5 and 3.1.3.6.b, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2.

      b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to both CEACs inoperable.

      c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "OFF" mode except during CEA motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3. At least once per 4 hours, all CEAs are verified fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, or as permitted by Specification 3.1.3.6.b, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in their group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - 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 affected reactor trip breakers within the next hour. The trip breakers associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

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TABLE 4.3-1

REACTOR PROTECTION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U (1)	N.A.
2. Linear Power Level – High	SFCP	SFCP (2,3,4)	SFCP	1,2
3. Logarithmic Power Level – High	SFCP	SFCP (4)	SFCP S/U (1)	1,2,3*,4*,5*
4. Pressurizer Pressure – High	SFCP	SFCP	SFCP	1,2
5. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1,2
6. Containment Pressure – High	SFCP	SFCP	SFCP	1,2
7. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1,2
8. Steam Generator Level – Low	SFCP	SFCP	SFCP	1,2
9. Local Power Density – High	SFCP	SFCP (2,4,5)	SFCP (6)	1,2
10. DNBR – Low	SFCP	SFCP (2,4,5,7)	SFCP (6)	1,2
11. Reactor Protection System Logic	N.A.	N.A.	SFCP	1,2,3*,4*,5*
12. Reactor Trip Breakers	N.A.	N.A.	SFCP	1,2,3*,4*,5*
13. Core Protection Calculators	SFCP	SFCP (2,4,5)	SFCP (6,9)	1,2
14. CEA Calculators	SFCP	SFCP	SFCP (6)	1,2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included):

a. Between 15% and 80% of RATED THERMAL POWER, compare the Linear Power Level, the CPC ΔT power, and the CPC nuclear power signals to the calorimetric calculation.

If any signal is within -0.5% to +10% of the calorimetric calculation, then do not calibrate except as required during initial power ascension following refueling.

If any signal is less than the calorimetric calculation by more than 0.5%, then adjust the affected signal(s) to within 0.0% to +10.0% of the calorimetric calculation.

If any signal is greater than the calorimetric calculation by more than 10%, then adjust the affected signal(s) to within 8% to 10% of the calorimetric calculation.

b. At or above 80% of RATED THERMAL POWER, compare the Linear Power Level, the CPC ΔT power, and CPC nuclear power signals to the calorimetric calculation. If any signal differs from the calorimetric calculation by an absolute difference of > 2%, then adjust the affected signal(s) to within ± 2% of the calorimetric calculation.

During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

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

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine or verify the shape annealing matrix elements used in the CPCs.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and, if necessary, adjust the CPC flow calibration addressable constant FC1 such that each CPC indicated flow is less than or equal to the measured flow rate.

(8) - Deleted

(9) - The CPC CHANNEL FUNCTIONAL TEST shall include the verification that the correct values of addressable constants are installed in each OPERABLE CPC.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	9
b. Containment Pressure - High	4	2	3	1, 2, 3	10,11
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	10,11
d. ESFAS Logic					
1. Matrix Logic	6	1	3	1, 2, 3	12
2. Initiation Logic	4	2	4	1, 2, 3, 4	9
e. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
2. CONTAINMENT SPRAY, MAIN STEAM, AND MAIN FEEDWATER ISOLATION (CSAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	9
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	10,11
c. ESFAS Logic					
1. Matrix Logic	6	1	3	1, 2, 3	12
2. Initiation Logic	4	2	4	1, 2, 3, 4	9
d. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13

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
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	9
b. Containment Pressure - High	4	2	3	1, 2, 3	10,11
c. ESFAS Logic					
1. Matrix Logic	6	1	3	1, 2, 3	12
2. Initiation Logic	4	2	4	1, 2, 3, 4	9
d. Automatic Actuation Logic	2	1	[illegible]	1, 2, 3, 4	13
4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	9
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	10,11
c. ESFAS Logic					
1. Matrix Logic	6	1	3	1, 2, 3	12
2. Initiation Logic	4	2	4	1, 2, 3, 4	9
d. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. CONTAINMENT COOLING (CCAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	9
b. Containment Pressure - High	4	2	3	1, 2, 3	10,11
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	10,11
d. ESFAS Logic					
1. Matrix Logic	6	1	3	1, 2, 3	12
2. Initiation Logic	4	2	4	1, 2, 3, 4	9
e. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
6. RECIRCULATION (RAS)					
a. Manual (TRIP Buttons)(c)	2 sets of 2	2 sets of 2	2 sets of 2	1, 2, 3, 4	9
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	10,11
c. ESFAS Logic					
1. Matrix Logic	6	1	3	1, 2, 3	12
2. Initiation Logic	4	2	4	1, 2, 3, 4	9
d. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13

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. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2/Bus	1/Bus	2/Bus	1,2,3	9,14
b. 460 volt Emergency Bus Undervoltage (Degraded Voltage)	2/Bus	2/Bus	2/Bus	1,2,3	14
8. EMERGENCY FEEDWATER (EFAS)					
a. Manual (Trip Switches)	2 sets of 2 per S/G	2 sets of 2 per S/G	2 sets of 2 per S/G	1,2,3	9
b. SG Level and Pressure (A/B) – Low and ΔP (A/B) – High	4/SG	2/SG	3/SG	1,2,3	10,11
c. SG Level (A/B) – Low and No S/G Pressure – Low Trip (A/B)	4/SG	2/SG	3/SG	1,2,3	10,11
d. ESFAS Logic					
1. Matrix Logic	6	1	3	1,2,3	12
2. Initiation Logic	4	2	4	1,2,3	9
e. Automatic Actuation Logic	2	1	2	1,2,3	13

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is below 400 psia; bypass shall be automatically removed before pressurizer pressure exceeds 500 psia.

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

(c) Remote manual not provided for RAS. These are local manuals at each ESF auxiliary relay cabinet.

ACTION STATEMENTS

ACTION 9 – With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 10 – With the number of channels OPERABLE one less than the Total Number of Channels, operation in the applicable MODES may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed for greater than 48 hours, the desirability of maintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM). The channel shall be returned to OPERABLE status prior to startup following the next COLD SHUTDOWN.

If an inoperable Steam Generator ΔP or RWT Level – Low channel is placed in the tripped condition, remove the inoperable channel from the tripped condition within 48 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	Functional Unit Bypassed
1. Containment Pressure – NR	Containment Pressure – High (RPS) Containment Pressure – High (ESFAS) Containment Pressure – High-High (ESFAS)
2. Steam Generator 1 Pressure	Steam Generator 1 Pressure – Low Steam Generator 1 ΔP (ESFAS 1) Steam Generator 2 ΔP (ESFAS 2)
3. Steam Generator 2 Pressure	Steam Generator 2 Pressure – Low Steam Generator 1 ΔP (ESFAS 1) Steam Generator 2 ΔP (ESFAS 2)

ACTION 10 (continued)

Process Measurement Circuit	Functional Unit Bypassed
4. Steam Generator 1 Level	Steam Generator 1 Level – Low Steam Generator 1 ΔP (EFAS 1)
5. Steam Generator 2 Level	Steam Generator 2 Level – Low Steam Generator 2 ΔP (EFAS 2)

ACTION 11 – With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, operation in the applicable MODES may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Containment Pressure – NR	Containment Pressure – High (RPS) Containment Pressure – High (ESFAS) Containment Pressure – High-High (ESFAS)
2. Steam Generator 1 Pressure	Steam Generator 1 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
3. Steam Generator 2 Pressure	Steam Generator 2 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
4. Steam Generator 1 Level	Steam Generator 1 Level – Low Steam Generator 1 ΔP (EFAS 1)
5. Steam Generator 2 Level	Steam Generator 2 Level – Low Steam Generator 2 ΔP (EFAS 2)

If an inoperable Steam Generator ΔP or RWT Level - Low channel is placed in the tripped condition, remove the inoperable channel from the tripped condition within 48 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours.

Operation in the applicable MODES may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent operation in the applicable MODES may continue if one channel is restored to OPERABLE status and the provisions of ACTION 10 are satisfied.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 12 – With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 13 – With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 14 – With the number of OPERABLE 460 volt Degraded Voltage (Functional Unit 7.b) channels one less than the Total Number of Channels or with both 4.16 kv Loss of Voltage (Functional Unit 7.a) channels inoperable on a single bus:

a. Immediately declare the affected diesel generator inoperable, and

b. Restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure – High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure – Low	≥ 1650 psia (1)	≥ 1618.9 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure – High-High	≤ 23.3 psia	≤ 23.490 psia
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure – High	≤ 18.3 psia	≤ 18.490 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure – Low	≥ 751 psia (2)	≥ 738.6 psia (2)
5. CONTAINMENT COOLING (CCAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure – High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure – Low	≥ 1650 psia	≥ 1618.9 psia
6. RECIRCULATION (RAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank – Low	6.0 ± 0.5% indicated level	between 5.111% and 6.889% indicated level
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage	(4)	3300.5 ± 49 volts with a 2.3 ± 0.3 second time delay
b. 460 volt Emergency Bus Undervoltage	(4)	429.6 ± 6.4 volts with an 8.0 ± 1.0 second time delay

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. EMERGENCY FEEDWATER (EFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator (A&B) Level – Low	≥ 22.2% (3)	≥ 21.5% (3)
c. Steam Generator ΔP– High (SG-A > SG-B)	≤ 90 psi	≤ 99.344 psi
d. Steam Generator ΔP– High (SG-B > SG-A)	≤ 90 psi	≤ 99.344 psi
e. Steam Generator (A&B) Pressure – Low	≥ 751 psia (2)	≥ 738.6 psia (2)

(1) Value may be decreased manually, to a minimum of ≥ 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip set-point is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed before pressurizer pressure exceeds 500 psia.

(2) Value may be decreased manually during a planned reduction in steam generator pressure, provided the margin between the steam generator pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower narrow range level instrument nozzles.

(4) The trip value for this function is listed in the surveillance test procedures. The trip value will ensure that adequate protection is provided when all the applicable calibration tolerances, channel uncertainties, and time delays are taken into account.

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TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. CONTAINMENT COOLING (CCAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Containment Pressure – High	SFCP	SFCP	SFCP	1,2,3
c. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1,2,3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3
6. RECIRCULATION (RAS)				
a. Manual (Trip Buttons) (a)	N.A.	N.A.	SFCP	N.A.
b. Refueling Water Tank – Low	SFCP	SFCP	SFCP	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3
7. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1,2,3
b. 460 volt Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1,2,3
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Switches)	N.A.	N.A.	SFCP	N.A.
b. SG Level and Pressure (A/B) – low and ΔP (A/B) – High	SFCP	SFCP	SFCP	1,2,3
c. SG Level (A/B) – Low and No Pressure – Low Trip (A/B)	SFCP	SFCP	SFCP	1,2,3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3

Table 4.3-2 (Continued)

TABLE NOTATION

(a) Remote manual not provided for RAS. These are local manuals at each ESF auxiliary relay cabinet.

(1) The logic circuits shall be tested manually at least once per 123 days.

INSTRUMENTATION

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 Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during 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 Monitor	1	Note 1	$\leq 1.5 \times 10^{-2}$ R/hr	$10^{-4} – 10^{1}$ R/hr	13
b. Containment High Range	2	1, 2, 3, & 4	Not Applicable	$1 – 10^{7}$ R/hr	18
2. PROCESS MONITORS					
a. Containment Purge and Exhaust Isolation	1	Note 3	$\leq$ 2 x background	$10 – 10^{6}$ cpm	16
b. Control Room Ventilation Intake Duct Monitors	2	Note 2	$\leq$ 2 x background	$10 – 10^{6}$ cpm	17,20,21

Note 1 – With fuel in the spent fuel pool or building.

Note 2 – MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

Note 3 – Applicable during:

a. PURGE of the Containment Building or,

b. Containment Building continuous ventilation operations when moving recently irradiated fuel assemblies or moving new fuel assemblies over recently irradiated fuel assemblies in the Containment Building.

TABLE NOTATION

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

ACTION 16 – With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, complete the following:

a. If moving recently irradiated fuel assemblies or moving new fuel assemblies over recently irradiated fuel assemblies within the Containment Building, secure the Containment Purge System or suspend the movement of recently irradiated fuel assemblies and movement of new fuel assemblies over recently irradiated fuel assemblies within the Containment Building.

b. If a Containment PURGE is in progress, secure the Containment Purge System.

c. If continuously ventilating the Containment Building, verify the associated SPING monitor operable or perform the applicable ACTION(s) of the Offsite Dose Calculation Manual; otherwise, secure the Containment Purge System.

ACTION 17 – In MODE 1, 2, 3, or 4, with no channels OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system (CREVS) in the recirculation mode of operation or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN in the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 18 – With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, (1) either restore the inoperable channel to OPERABLE status within 7 days or (2) prepare and submit a Special Report to the NRC within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

ACTION 19 – DELETED

ACTION 20 – In MODE 1, 2, 3, or 4 with the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, within 7 days restore the inoperable channel to OPERABLE status or initiate and maintain the CREVS in the recirculation mode of operation. Otherwise, be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN in the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 21 – During movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies with one or two channels inoperable, immediately place one OPERABLE CREVS train in the emergency recirculation mode or immediately suspend the movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area Monitor	SFCP	SFCP	SFCP	Note 1
b. Containment High Range	SFCP	SFCP Note 4	SFCP	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Purge and Exhaust Isolation	Note 2	Note 3	Note 3	In accordance with applicable Notes
b. Control Room Ventilation Intake Duct Monitors	SFCP	SFCP	SFCP Note 6	Note 5

Note 1 – With fuel in the spent fuel pool or building.

Note 2 – Within 8 hours prior to initiating Containment PURGE operations and in accordance with the Surveillance Frequency Control Program during Containment PURGE or continuous ventilation operations.

Note 3 – Within 31 days prior to initiating Containment PURGE operations and in accordance with the Surveillance Frequency Control Program during Containment continuous ventilation operations when moving recently irradiated fuel assemblies or moving new fuel assemblies over recently irradiated fuel assemblies in the Containment Building.

Note 4 – Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

Note 5 – MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

Note 6 - When the Control Room Ventilation Intake Duct Monitor is placed in an inoperable status solely for performance of this Surveillance, entry into associated ACTIONS may be delayed up to 3 hours.

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the number of OPERABLE remote shutdown monitoring channels less than required, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK and CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program. The logarithmic neutron instrumentation, the startup channel instrumentation, and the reactor trip breaker indication are excluded from CHANNEL CALIBRATION.

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INSTRUMENTATION

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring (PAM) instrumentation channels for each Function shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

In accordance with Table 3.3-10.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE in accordance with Table 4.3-10.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

FUNCTION		REQUIRED CHANNELS	ACTION
1.	Penetration Flow Path Containment Isolation Valve Position	2 per Penetration Flow Path[(a)(b)]	1, 2
2.	Containment Pressure (Wide Range)	2	1, 2
3.	Pressurizer Pressure (Wide Range)	2	1, 2
4.	Pressurizer Level	2	1, 2
5.	Steam Generator (SG) Pressure	2 per SG	1, 2
6.	SG Water Level (Wide Range)	2 per SG	1, 2
7.	Refueling Water Tank Water Level	2	1, 2
8.	Containment Water Level (Wide Range)	2	1, 2
9.	Emergency Feedwater Flow Rate	2 per SG	1, 2
10.	Reactor Coolant System Hot Leg Temperature (Narrow Range)	2 per Loop	1, 2
11.	Reactor Coolant System Hot Leg Temperature (Wide Range)	2 per Loop	1, 2
12.	High Pressure Safety Injection Flow Rate	1 per Train	1, 2
13.	Core Exit Thermocouples (CETs) – Quadrant 1	2	1, 2
14.	CETs – Quadrant 2	2	1, 2
15.	CETs – Quadrant 3	2	1, 2
16.	CETs – Quadrant 4	2	1, 2
17.	Reactor Vessel Level Monitoring System (RVLMS)	2	1, 2, 3

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed Control Room indication channel.

TABLE 3.3-10 (cont'd)

POST-ACCIDENT MONITORING INSTRUMENTATION

ACTIONs[1]

1) With one or more Table 3.3-10 Functions with one required channel inoperable, restore the required channel to OPERABLE status within 30 days. If not restored to an OPERABLE status within 30 days, immediately initiate action in accordance with Specification 6.6.4.

2)[2] With one or more Table 3.3-10 Functions with no required channel OPERABLE, restore at least one inoperable channel to OPERABLE status within 7 days, or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

3) With no required Table 3.3-10 RVLMS Function channel OPERABLE and repair is not feasible without shutting down, immediately initiate action in accordance with Specification 6.6.4.

Note 1 – Separate ACTION entry is allowed for each Table 3.3-10 Function.

Note 2 – Action 2 is applicable to the RVLMS Function only when repair is feasible without shutting down. Where RVLMS channel repair is not feasible, Action 3 shall be applicable.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTION		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Penetration Flow Path Containment Isolation Valve Position	SFCP	SFCP
2.	Containment Pressure (Wide Range)	SFCP	SFCP
3.	Pressurizer Pressure (Wide Range)	SFCP	SFCP
4.	Pressurizer Level	SFCP	SFCP
5.	Steam Generator (SG) Pressure	SFCP	SFCP
6.	SG Water Level (Wide Range)	SFCP	SFCP
7.	Refueling Water Tank Water Level	SFCP	SFCP
8.	Containment Water Level (Wide Range)	SFCP	SFCP
9.	Emergency Feedwater Flow Rate	SFCP	SFCP
10.	Reactor Coolant System Hot Leg Temperature (Narrow Range)	SFCP	SFCP
11.	Reactor Coolant System Hot Leg Temperature (Wide Range)	SFCP	SFCP
12.	High Pressure Safety Injection Flow Rate	SFCP	SFCP
13.	Core Exit Thermocouples (CETs) – Quadrant 1	SFCP	SFCP
14.	CETs – Quadrant 2	SFCP	SFCP
15.	CETs – Quadrant 3	SFCP	SFCP
16.	CETs – Quadrant 4	SFCP	SFCP
17.	Reactor Vessel Level Monitoring System (RVLMS)	SFCP	SFCP

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

APPLICABILITY: MODES 1 and 2. *

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

* See Special Test Exception 3.10.3.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

b. At least one of the above Reactor Coolant Loops shall be in operation.*

APPLICABILITY: MODE 3.

ACTION:

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

b. With no reactor coolant loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 and immediately initiate corrective action to return the required loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

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

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.
2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.
3. Shutdown Cooling Loop (A) #.
4. Shutdown Cooling Loop (B) #.

b. At least one of the above coolant loops shall be in operation.*

APPLICABILITY: Modes 4 and 5.

ACTION:

a. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible and initiate action to make at least one steam generator available for decay heat removal via natural circulation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

b. With no coolant loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 or LCO 3.1.1.2, as applicable, and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required shutdown cooling loop(s) shall be determined OPERABLE per the INSERVICE TESTING PROGRAM.

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

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥ 23% indicated level in accordance with the Surveillance Frequency Control Program.

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

---

* 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 introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 or LCO 3.1.1.2, as applicable, and (2) core outlet temperature is maintained at least 10 °F below saturation temperature.

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

REACTOR COOLANT SYSTEM

SAFETY VALVES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA ± 3%*.

APPLICABILITY: MODE 4 with Tc > 220 °F.

ACTION:

With no pressurizer code safety valve OPERABLE, reduce Tc to ≤ 220 °F within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ± 1% tolerance band, the setting shall be adjusted to within ± 1% of the lift setting shown.

REACTOR COOLANT SYSTEM

SAFETY VALVES – OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting 2500 psia ± 3%*.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

b. The provisions of specification 3.0.4.c may be applied and the requirements of ACTION "a" suspended for one valve at a time for up to 18 hours for entry into and during operation in MODE 3 for the purpose of setting the pressurizer code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

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* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ± 1% tolerance band, the setting shall be adjusted to within ± 1% of the lift setting shown.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of ≤ 910 cubic feet (equivalent to ≤ 82% of wide range indicated level) and both pressurizer proportional heater groups shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(a) With the pressurizer inoperable due to water volume ≥ 910 cubic feet, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

(b) With the pressurizer inoperable due to a single proportional heater group having less than a 150 KW capacity, restore the inoperable proportional heater group to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within 12 hours.

(c) With the pressurizer inoperable due to both proportional heater groups being inoperable for any reason (Note 1), restore at least one proportional heater group to OPERABLE status within 24 hours, or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.4.2 The pressurizer proportional heater groups shall be determined to be OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that the summed power consumption of the two proportional heater groups is ≥ 150 KW.

Note 1: Action (d) is not applicable when the second group of required pressurizer heaters is intentionally made inoperable.

STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 a. SG tube integrity shall be maintained, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Note: ACTIONS may be entered separately for each SG tube.

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

1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and
2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

b. If the required ACTION and Allowed Outage Time of ACTION a above cannot be met or SG tube integrity cannot be maintained, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. One containment sump level monitor

b. One containment atmosphere particulate radioactivity monitor, and

c. One containment atmosphere gaseous radioactivity monitor.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one or more containment atmosphere radioactivity monitor(s) inoperable, operation may continue for up to 30 days for each inoperable monitor provided:

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

   2. a Reactor Coolant System water inventory balance is performed at least once per 24 hours in accordance with Surveillance Requirement 4.4.6.2.1.a;*

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

b. With the containment sump level monitor inoperable, operation may continue for up to 30 days provided a Reactor Coolant System water inventory balance is performed at least once per 24 hours in accordance with Surveillance Requirement 4.4.6.2.1.a;* otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the containment sump level monitor inoperable and one containment atmosphere radioactivity monitor inoperable, operation may continue for up to 30 days for each inoperable monitor provided a Reactor Coolant System water inventory balance is performed at least once per 24 hours in accordance with Surveillance Requirement 4.4.6.2.1.a;* otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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* Not required until 12 hours after establishment of steady state conditions.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection instrumentation shall be demonstrated OPERABLE by:

a. Performing a CHANNEL CHECK of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program.

b. Performing a CHANNEL CHECK of the containment sump level monitor in accordance with the Surveillance Frequency Control Program.

c. Performing a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program.

d. Performing a CHANNEL CALIBRATION of the containment sump level monitor in accordance with the Surveillance Frequency Control Program.

e. Performing a CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

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 (SG),

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

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any 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 PRESSURE BOUNDARY LEAKAGE not within limit, isolate affected component, pipe, or vessel from the Reactor Coolant System by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve 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 operational leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and primary to secondary leakage, 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.

d. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate (Note 1) the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two valves* in each high pressure line having a non-functional valve and be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Note 1: Enter applicable ACTION(s) for systems(s) made inoperable by an inoperable pressure isolation valve.

* These valves may include check valves for which the leakage rate has been verified, manual valves or automatic valves. Manual and automatic valves shall be tagged as closed to preclude inadvertent valve opening.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakage, except for primary to secondary leakage, shall be demonstrated to be within each of the above limits by performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program during steady state operation except when operating in the shutdown cooling mode*.

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

4.4.6.2.3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4.6-1 shall be demonstrated OPERABLE by individually verifying leakage to be within its limit:

a. Prior to entering MODE 2 after each refueling outage,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months, and

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

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

TABLE 3.4.6-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES [(a)(b)(c)]
(CHECK VALVES)

System	Check Valve No.
High-Pressure Safety Injection	
Loop A, cold leg	2SI-15A 2SI-13A
Loop B, cold leg	2SI-15B 2SI-13B
Loop C, cold leg	2SI-15C 2SI-13C
Loop D, cold leg	2SI-15D 2SI-13D
Low-Pressure Safety Injection	
Loop A, cold leg	2SI-14A
Loop B, cold leg	2SI-14B
Loop C, cold leg	2SI-14C
Loop D, cold leg	2SI-14D

NOTES

(a) Maximum Allowable Leakage (each valve):

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

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

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

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

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

ACTION

Note: The provisions of Specification 3.0.4.c are applicable to ACTION a and b.

a. With the DOSE EQUIVALENT I-131 not within limit:

   1. Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gm once every 4 hours, and

   2. Restore DOSE EQUIVALENT I-131 within limit within 48 hours.

b. With the DOSE EQUIVALENT XE-133 not within limit, restore DOSE EQUIVALENT XE-133 within limit within 48 hours.

c. With the requirements of ACTION a and/or b not met, or with DOSE EQUIVALENT I-131 > 60 μCi/gm, be in at least HOT STANDBY in 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1 Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 3100 μCi/gm in accordance with the Surveillance Frequency Control Program.*

4.4.8.2 Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm:*

   a. in accordance with the Surveillance Frequency Control Program, and

   b. between 2 and 6 hours after THERMAL POWER change of ≥ 15% RATED THERMAL POWER within a 1 hour period.

* Only required to be performed in MODE 1.

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REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. A maximum heatup of 50 °F, 60 °F, 70 °F or 80 °F in any one hour period in accordance with Figure 3.4-2A.

b. A maximum cooldown rate of 100 °F per hour (constant) or 50 °F in any half hour period (step) for RCS cold leg temperatures between 60 °F and 560 °F.

c. A maximum temperature change of ≤ 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:

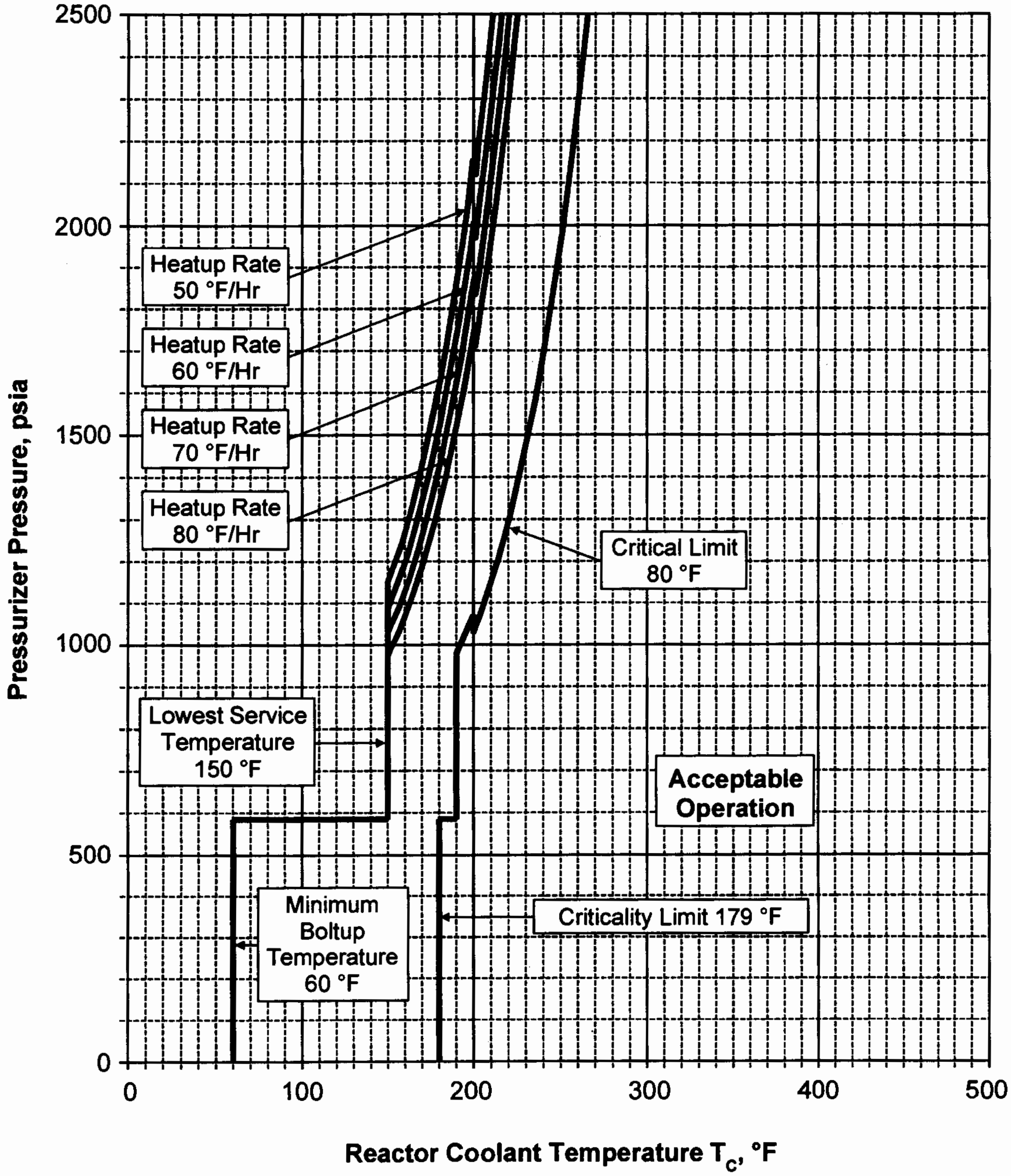
With any of the above limits exceeded, restore the temperature and/or pressure to within the acceptable region of the applicable curve within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant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

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

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 shown in SAR Table 5.2-12. The results of these examinations shall be used to update Figures 3.4-2A, 3.4-2B and 3.4-2C.

Figure 3.4-2A

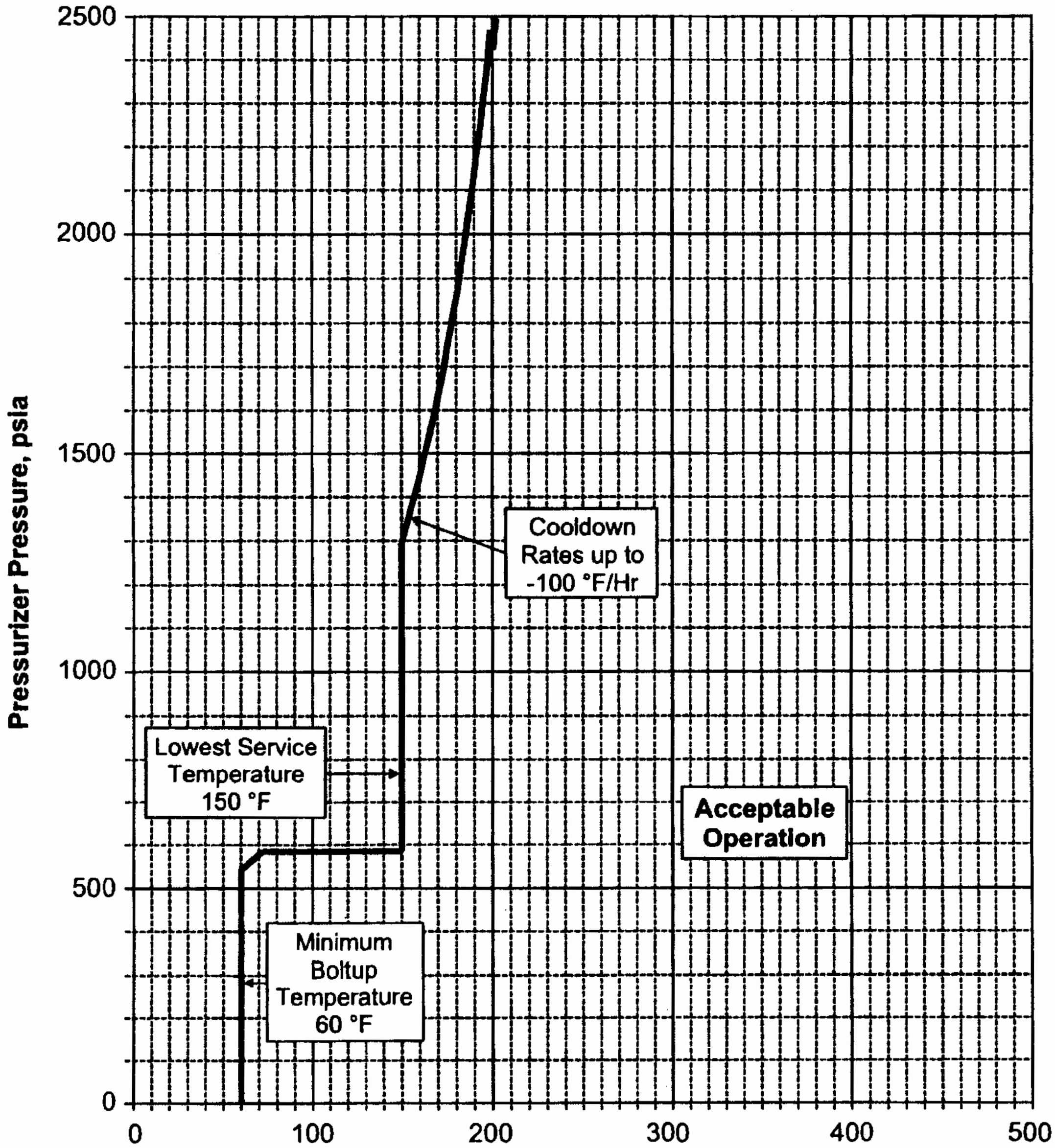
HEATUP CURVE – 54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



(Curves do not include margins for instrument uncertainties)

Figure 3.4-2B

COOLDOWN CURVE – 54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS

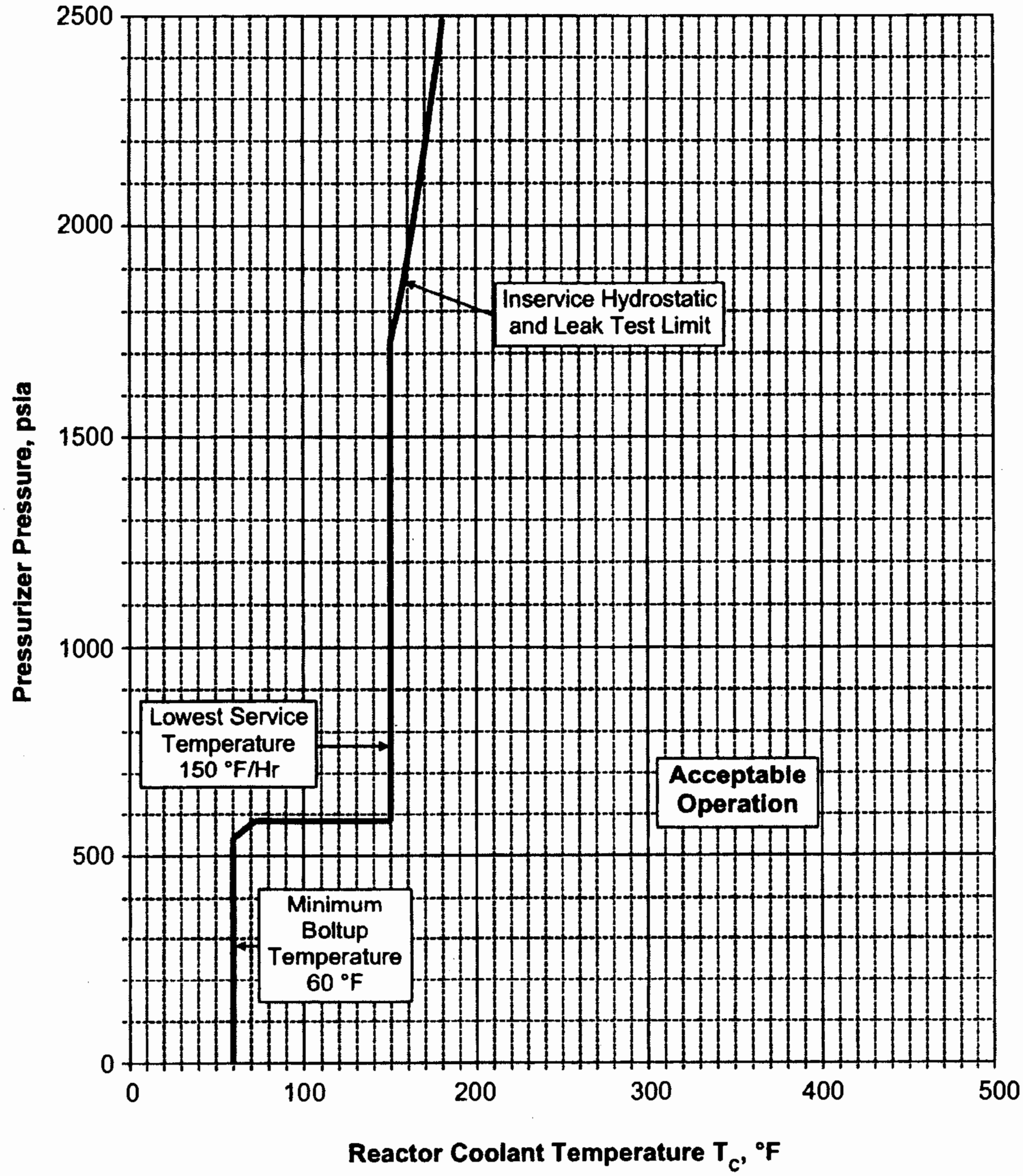


Reactor Coolant Temperature T_c, °F

(Curves do not include margins for instrument uncertainties)

Figure 3.4-2C

INSERVICE HYDROSTATIC TEST CURVE – 54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



(Curves do not include margins for instrument uncertainties)

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor coolant system vent path consisting of at least two valves in series shall be OPERABLE at each of the following locations:

1. Reactor Vessel Head

2. Pressurizer Steam Space (RCS High Point Vents)

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

ACTION:

a. With less than one vent path from each of the locations OPERABLE, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path(s) is maintained closed; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both vent paths 1 and 2 above inoperable, restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor coolant system vent path shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying flow through the reactor coolant vent system vent paths.

REACTOR COOLANT SYSTEM

LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.12 The LTOP system shall be OPERABLE with each SIT isolated that is pressurized to ≥ 300 psig, and a maximum of one HPSI pump capable of injecting into the RCS and:

a. Two LTOP relief valves with a lift setting of ≤ 430 psig, or

b. The Reactor Coolant System depressurized with an RCS vent path ≥ 6.38 square inches.

APPLICABILITY: MODE 4 with T_C ≤ 220°F, MODE 5, MODE 6 with reactor vessel head in place.*

ACTION:

NOTE: Specification 3.0.4.b is not applicable to LTOP relief valves when entering Mode 4.

a. With one LTOP relief valve inoperable in MODE 4, restore the inoperable valve to OPERABLE status within 7 days or depressurize and vent the RCS through a ≥ 6.38 square inch vent path within the next 8 hours.

b. With one LTOP relief valve inoperable in MODE 5 or 6, restore the inoperable relief valve to OPERABLE status within 24 hours or depressurize and vent the RCS through a ≥ 6.38 square inch vent path within the next 8 hours.

c. With both LTOP relief valves inoperable, depressurize and vent the RCS through a ≥ 6.38 square inch vent path within 8 hours.

d. With a SIT not isolated and pressurized to ≥ 300 psig, isolate the affected SIT within 1 hour. If the affected SIT is not isolated within 1 hour, either:

(1) Depressurize the SIT to < 300 psig within the next 12 hours, or

(2) Increase cold leg temperature to > 220°F within the next 12 hours.

e. With more than one HPSI pump capable of injecting into the RCS, immediately initiate action to verify a maximum of one HPSI pump capable of injecting into the RCS.

* - when starting the first reactor coolant pump, the pressurizer water volume will be < 910 ft^3.

4.4.12.1 Verify both sets of LTOP relief valve isolation valves are open in accordance with the Surveillance Frequency Control Program when the LTOP relief valves are being used for overpressure protection.

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

4.4.12.3 Verify that each SIT is isolated, when required, in accordance with the Surveillance Frequency Control Program.

4.4.12.4 No additional LTOP relief valve Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

** Except when the vent path is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify this valve is open in accordance with the Surveillance Frequency Control Program.

SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open,

b. A contained borated water volume of between 1413 and 1539 cubic feet (equivalent to an indicated level between 80.1% and 87.9%, respectively),

c. Between 2200 and 3000 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 624 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, due to boron concentration not within limits, restore the boron concentration to within limits within 72 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

b. With one safety injection tank inoperable due to inability to verify level or pressure, restore the SIT to OPERABLE status within 72 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

c. With one safety injection tank inoperable for reasons other than ACTION a or b, restore the SIT to OPERABLE status within 24 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

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

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

2. Verifying that each safety injection tank isolation valve (2CV-5003-1, 2CV-5023-1, 2CV-5043-2, and 2CV-5063-2) is open.

* With pressurizer pressure ≥ 700 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of ≥ 5% of indicated tank level that is not the result of addition from the RWT, by verifying the boron concentration of the safety injection tank solution.

c. In accordance with the Surveillance Frequency Control Program when the RCS pressure is above 2000 psia, by verifying that power to the isolation valve operator is removed by maintaining the motor circuit breaker open under administrative control.

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

   1. When the RCS pressure exceeds 700 psia, and

   2. Upon receipt of a safety injection test signal.

EMERGENCY CORE COOLING SYSTEMS

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

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each sub-system comprised of:

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

b. One OPERABLE low-pressure safety injection (LPSI) train, and

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

APPLICABILITY: MODES 1, 2 and 3 with pressurizer pressure ≥ 1700 psia.

ACTION:

a. With one ECCS subsystem inoperable due to an inoperable LPSI train, restore the inoperable train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program; otherwise, be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than "a" above and 100% of ECCS flow equivalent to a single OPERABLE HPSI and LPSI train is available, restore the inoperable train(s) to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

c. With less than 100% ECCS flow equivalent to either the HPSI or LPSI trains within both ECCS subsystems, restore at least one HPSI train and one LPSI train to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

d. 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 NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the 2CV-5101-1 and 2CV-5102-2 valve operators removed:

Valve Number	Valve Function	Valve Position
2CV-5101-1	HPSI Hot Leg Injection Isolation	Closed
2CV-5102-2	HPSI Hot Leg Injection Isolation	Closed
2BS-26	RWT Return Line	Open

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

c. DELETED

d. DELETED

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and RAS test signals.

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

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to the INSERVICE TESTING PROGRAM:

1. High-Pressure Safety Injection pump ≥ 1360.4 psid with 90 °F water.

2. Low-Pressure Safety Injection pump ≥ 156.25 psid with 90 °F water.

g. In accordance with the Surveillance Frequency Control Program by verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

LPSI System
Valve Number

a. 2CV-5037-1

b. 2CV-5017-1

c. 2CV-5077-2

d. 2CV-5057-2

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

HPSI System – Single Pump

The sum of the injection line flow rates, excluding the highest flow rate is greater than or equal to 570 gpm.

LPSI System – Single Pump

a. Injection Leg 1, ≥ 1059 gpm

b. Injection Leg 2, ≥ 1059 gpm

c. Injection Leg 3, ≥ 1059 gpm

d. Injection Leg 4, ≥ 1059 gpm

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS – T_{avg} < 300°F

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE high-pressure safety injection pump, and

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

APPLICABILITY: MODES 3* and 4.

ACTION:

NOTE: Specification 3.0.4.b is not applicable to ECCS subsystem when entering Mode 4.

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 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 NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

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

* With pressurizer pressure < 1700 psia.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. An available borated water volume of between 384,000 and 503,300 gallons

b. Between 2500 and 3000 ppm of boron,

c. A minimum solution temperature of 40 °F, and

d. A maximum solution temperature of 110 °F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

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

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

2. Verifying the boron concentration of the water.

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

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 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. In accordance with the Surveillance Frequency Control Program by verifying that each containment isolation manual valve and blind flange (Note 1) that is located outside containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative control as permitted by Specification 3.6.3.1.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals in accordance with the Containment Leakage Rate Testing Program.

d. Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days by verifying each containment isolation manual valve and blind flange (Note 1) that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls as permitted by Specification 3.6.3.1.

Note 1: Valves and blind flanges in high radiation areas may be verified by use of administrative means.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment leakage rate exceeding the acceptance criteria of the Containment Leakage Rate Testing Program, within 1 hour, restore leakage to within limits or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable in one or more containment air locks[1,2]:

   1. Verify that at least the OPERABLE air lock door is closed in the affected air lock within one hour and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed[3].

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

b. With the containment air lock interlock inoperable in one or more containment air locks[1]:

   1. Verify that at least one OPERABLE air lock door is closed in the affected air lock within one hour and restore the inoperable air lock interlock to OPERABLE status within 24 hours or lock an OPERABLE air lock door closed[4].

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

c. With one or more air locks inoperable for reasons other than those addressed in ACTION a. or b.:

   1. Immediately initiate action to evaluate overall containment leakage per LCO 3.6.1.2.

   2. Verify that at least one door in the affected air lock is closed within one hour and restore the affected air lock to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program.

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

---

[1] Separate ACTION entry is allowed for each air lock.

[2] With both air locks inoperable, entry and exit is permissible for seven days under administrative controls.

[3] Entry and exit is permissible to perform repairs on the affected air lock components.

[4] Entry and exit is permissible under the control of a dedicated individual.

SURVEILLANCE REQUIREMENTS

4.6.1.3.1 Each containment air lock shall be demonstrated OPERABLE as specified in the Containment Leakage Rate Testing Program[5,6].

4.6.1.3.2 Each containment air lock interlock shall be demonstrated OPERABLE by testing the air lock interlock mechanism in accordance with the Surveillance Frequency Control Program[7].

---

[5] Leakrate results shall also be evaluated against the acceptance criteria of specification 3.6.1.2.

[6] An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

[7] This surveillance requirement is only required to be performed upon entry or exit through the associated containment air lock.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE AND AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 The combination of containment internal pressure and average air temperature shall be maintained within the region of acceptable operation shown on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

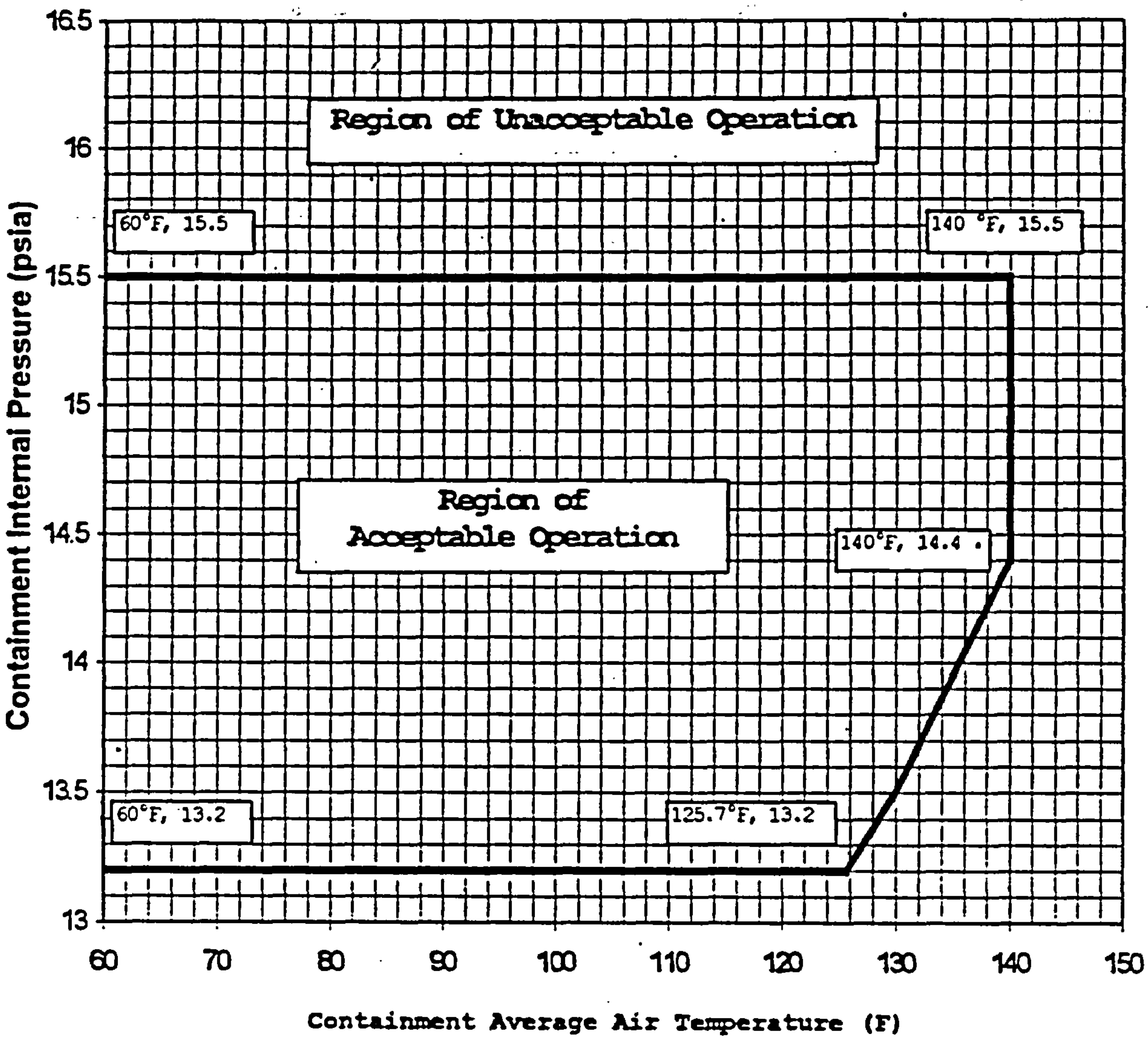
With the point defined by the combination of containment internal pressure and average air temperature outside the region of acceptable operation shown on Figure 3.6-1, restore the combination of containment internal pressure and average air temperature to within the above limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure and average air temperature shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program. The containment average air temperature shall be the temperature of the air in the containment HVAC common return air duct upstream of the fan/cooler units.

FIGURE 3.6-1

CONTAINMENT INTERNAL PRESSURE VS. AVERAGE AIR TEMPERATURE



NOTE: Instrument Error is not Included on Curve

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

If the containment is not OPERABLE, restore the structural integrity 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

4.6.1.5. Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.6 The containment purge supply and exhaust isolation valves shall be closed and handswitch keys removed.

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

ACTION:

With one or more containment purge supply and/or exhaust isolation valves not closed with the handswitch keys removed, place the valve(s) in the closed position with handswitch keys(s) removed 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 The containment purge supply and exhaust isolation valves shall be determined closed in accordance with the Surveillance Frequency Control Program.

3/4.6.2 DEPRESSURIZATION, COOLING, AND pH CONTROL SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a Containment Spray Actuation Signal (CSAS) and automatically transferring suction to the containment sump on a Recirculation Actuation Signal (RAS). Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With both containment spray systems inoperable (Note 1):

   1. Within 1 hour verify both CREVS trains are OPERABLE, and

   2. Restore at least one containment spray system to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program.

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

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

   1. Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.

   2. Verifying that the system piping is full of water from the RWT to at least elevation 505' (equivalent to > 12.5% indicated narrow range level) in the risers within the containment.

b. Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head when tested pursuant to the INSERVICE TESTING PROGRAM.

Note 1: ACTION b is not applicable when the second containment spray system is intentionally made inoperable.

SURVEILLANCE REQUIREMENTS (Continued)

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

   1. Verifying that each automatic valve in the flow path actuates to its correct position on CSAS and RAS test signals.

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

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

d. Verify each spray nozzle is unobstructed following maintenance which could result in nozzle blockage.

CONTAINMENT SYSTEMS

CONTAINMENT SUMP BUFFERING AGENT

LIMITING CONDITION FOR OPERATION

3.6.2.2 The buffering agent baskets shall contain ≥ 308 ft^3 of sodium tetraborate (NaTB) decahydrate.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the buffering agent not within limits, restore the buffering agent to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and be in at least HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The buffering agent shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the buffering agent baskets contain ≥ 308 ft^3 of NaTB decahydrate.

b. In accordance with the Surveillance Frequency Control Program by verifying that a sample from the buffering agent baskets provides adequate pH adjustment of borated water.

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-effective as of its date of issuance to be implemented DEC 2 3 1998
prior to the facility's restart from refueling outage 2R13

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Two independent containment cooling groups shall be OPERABLE with two operational cooling units in each group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION[1]:

a. With one group of the above required containment cooling units inoperable and both containment spray systems OPERABLE, restore the inoperable group of cooling units to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program.

b. With two groups of the above required containment cooling units inoperable and both containment spray systems OPERABLE, restore at least one group of cooling units to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program. Restore both above required groups of cooling units to OPERABLE status within 7 days, or in accordance with the Risk Informed Completion Time Program, of initial loss.

c. With one group of the above required containment cooling units inoperable and one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program. Restore the inoperable group of containment cooling units to OPERABLE status within 7 days, or in accordance with the Risk Informed Completion Time Program, of initial loss.

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

Note 1: The containment spray systems may be considered OPERABLE with respect to ACTIONs a, b, and c above if solely inoperable due to containment accident generated and transported debris exceeding the analyzed limits and LCO 3.6.4.1, ACTION a, is being met.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling group shall be demonstrated OPERABLE:

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

1. Verifying that service water flow rate to the group of cooling units is ≥ 1250 gpm and that each group has two operable fans.

2. Addition of a biocide to the service water during the surveillance in 4.6.2.3.a.1 above, whenever service water temperature is between 60 °F and 80 °F.

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

1. Starting (unless already operating) each operational cooling unit from the control room.

2. Verifying that each operational cooling unit operates for at least 15 minutes.

c. In accordance with the Surveillance Frequency Control Program by verifying that each cooling unit starts automatically on a CCAS test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Note: Enter applicable ACTION(s) for system(s) made inoperable by containment isolation valves.

With one or more isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours or in accordance with the Risk Informed Completion Time Program either:

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

b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration by use of at least one closed manual valve or blind flange; or

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

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 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.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position.

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

4.6.3.1.4 The containment purge supply and exhaust isolation valves shall be demonstrated OPERABLE as specified in the Containment Leakage Rate Testing Program.

CONTAINMENT SYSTEMS

CONTAINMENT SUMP

LIMITING CONDITION FOR OPERATION

3.6.4.1 The containment sump shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the containment sump inoperable due to containment accident generated and transported debris exceeding the analyzed limits, entry into the applicable ACTION(s) of LCO 3.5.2, "ECCS Subsystems – $T_{avg} \geq 300$ °F," LCO 3.5.3, "ECCS Subsystems – $T_{avg} < 300$ °F," and LCO 3.6.2.1, "Containment Spray System," is not required provided:

   1. Action is initiated immediately to mitigate containment accident generated and transported debris, and

   2. SR 4.4.6.2.1.a is performed once every 24 hours, and

   3. The containment sump is restored to OPERABLE status within 90 days.

b. With the containment sump inoperable for reasons other than ACTION a:

   1. Immediately enter the applicable ACTIONS of LCO 3.5.2, "ECCS Subsystems – $T_{avg} \geq 300$ °F" and LCO 3.5.3, "ECCS Subsystems – $T_{avg} < 300$ °F," for emergency core cooling trains made inoperable by the containment sump, and

   2. Immediately enter the applicable ACTIONS of LCO 3.6.2.1, "Containment Spray System," for containment spray trains made inoperable by the containment sump, and

   3. Restore the containment sump to OPERABLE status within 72 hours.

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

SURVEILLANCE REQUIREMENTS

4.6.4.1.1 Verify, by visual inspection, that the containment sump does not show structural damage, abnormal corrosion, or debris blockage in accordance with the Surveillance Frequency Control Program.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-5.

APPLICABILITY: MODES 1, 2 and 3*

ACTION:

MODES 1 and 2

With one or more main steam line code safety valves inoperable, operation in MODES 1 and 2 may proceed provided that within 4 hours, power is reduced to less than or equal to the applicable percent of RATED THERMAL POWER as listed in Table 3.7-1 and within 12 hours, the Linear Power Level – High trip setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

MODE 3

With one or more main steam line code safety valves inoperable, operation in MODE 3 may proceed provided that at least 2 main steam line code safety valves are OPERABLE on each steam generator; otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

* Except that during hydrostatic testing in Mode 3, eight of the main steam line code safety valves may be gagged and two (one on each header) may be reset for the duration of the test to allow the required pressure for the test to be attained. The Reactor Trip Breakers shall be open for the duration of the test.

TABLE 3.7-1

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

Number of Inoperable Safety Valves	Maximum Allowable Linear Power Level And High Trip Setpoint (Percent of RATED THERMAL POWER)
1 Valve Inoperable	79% (except as allowed by Figure 3.7-1)
1 Valve Inoperable on Each Header	71% (except as allowed by Figure 3.7-1)
Maximum of 2 Valves Inoperable on Each Header	43.0
Maximum of 3 Valves Inoperable on Each Header	25.0

FIGURE 3.7-1

Maximum High Linear Power Level And Trip Setpoint
Versus MTC

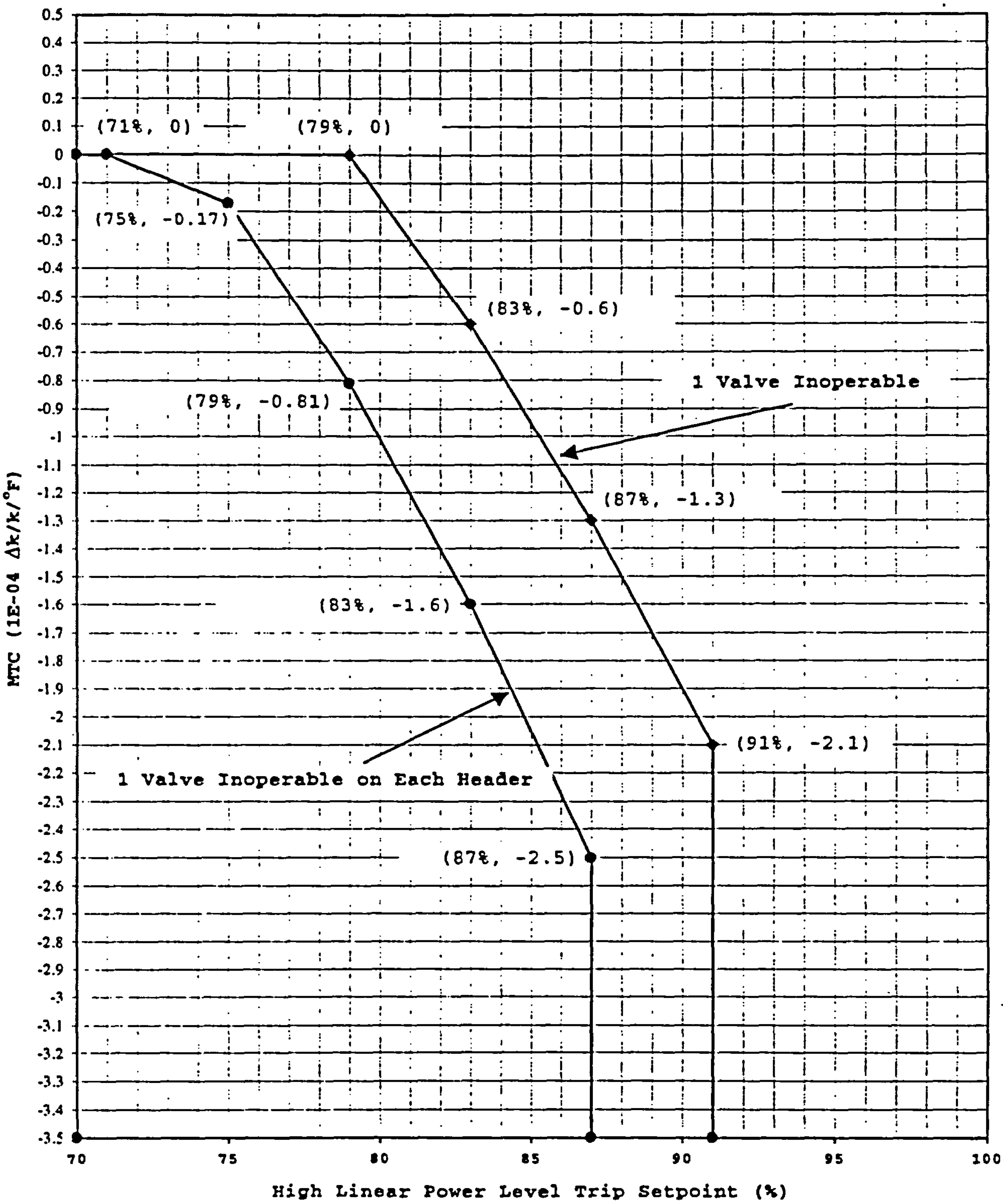

TABLE 3.7-5

STEAM LINE SAFETY VALVES

	VALVE NUMBER		LIFT SETTING (± 3%) *
	Line No. 1	Line No. 2	
a.	2PSV-1002	2PSV-1052	1078 psig
b.	2PSV-1003	2PSV-1053	1105 psig
c.	2PSV-1004	2PSV-1054	1105 psig
d.	2PSV-1005	2PSV-1055	1132 psig
e.	2PSV-1006	2PSV-1056	1132 psig

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ± 1% tolerance band, the setting shall be adjusted to within ± 1% of the lift setting shown.

PLANT SYSTEMS

EMERGENCY FEEDWATER (EFW) SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two EFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS:[1]

NOTE 1: Specification 3.0.4.b is not applicable.

NOTE 2: Only applicable if MODE 2 has not been entered following refueling.

NOTE 3: Not applicable when the turbine-driven EFW train is inoperable solely due to one inoperable steam supply.

NOTE 4: LCO 3.0.3 and all other LCO ACTIONS requiring MODE changes are suspended until one EFW train is restored to OPERABLE status.

a. With the turbine-driven EFW train inoperable in MODE 3 following refueling[2], OR with the turbine-driven EFW train inoperable due to one inoperable steam supply, restore the turbine-driven EFW train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program.

b. With one EFW train inoperable for reasons other than ACTION a, restore the inoperable train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program.

c. With the turbine-driven EFW train inoperable due to one inoperable steam supply AND the motor-driven EFW train inoperable, restore either the steam supply to the turbine-driven train OR the motor-driven EFW train to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program.

d. With ACTION a, b, or c not met, be in HOT SHUTDOWN within the next 12 hours.

e. With both EFW trains inoperable, immediately initiate action to restore one EFW train to an OPERABLE status.[3,4]

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each EFW pump shall be demonstrated OPERABLE:

a In accordance with the Surveillance Frequency Control Program by:

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

b. In accordance with the INSERVICE TESTING PROGRAM by:

1. Verifying the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head. This surveillance requirement is not required to be performed for the turbine-driven EFW pump until 24 hours after exceeding 700 psia in the steam generators.

c In accordance with the Surveillance Frequency Control Program by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on actual or simulated MSIS and EFAS.

2. Verifying each EFW pump starts automatically on an actual or simulated EFAS.

d. By verifying proper alignment of the required EFW flow paths by verifying flow from the condensate storage tank to each steam generator. This SR is required to be verified prior to entering MODE 2 whenever plant has been in MODES 4, 5, 6, or defueled for > 30 days.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 At least one condensate storage tank (CST) shall be OPERABLE with a minimum contained water volume of either:

a. 160,000 gallons in either 2T41A or 2T41B, or

b. A minimum of 267,000 gallons of water is available in condensate storage tank, T41B, when required for both units. A minimum of 160,000 gallons of water is available in T41B when only required for Unit 2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the required condensate storage tank inoperable, within 4 hours either:

a. Restore at least one CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore at least one condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The above required condensate storage tank shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program 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 in accordance with the Surveillance Frequency Control Program by verifying that at least one service water loop is operating and that the service water system - emergency feedwater system isolation valves are either open or OPERABLE whenever the service water system is the supply source for the emergency feedwater pumps.

ACTIVITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	In accordance with the Surveillance Frequency Control Program
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination is greater than 10% of the allowable iodine limit. b) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination is below 10% of the allowable iodine limit.

PLANT SYSTEMS

MAIN STEAM ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed provided the isolation valve is maintained closed; otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam isolation valve shall be demonstrated OPERABLE by verifying full closure within 3 seconds when tested pursuant to the INSERVICE TESTING PROGRAM.

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PLANT SYSTEMS

3/4.7.3 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Notes:

1. Enter applicable ACTION(s) of LCO 3.8.1.1, "AC Sources – Operating," for diesel generator made inoperable by service water system.

2. Enter applicable ACTION(s) of LCO 3.4.1.3, "Reactor Coolant System – Shutdown," if a required shutdown cooling loop is made inoperable by service water system.

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

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two service water loops shall be demonstrated OPERABLE:

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

b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on CCAS, MSIS and RAS test signals.

3/4.7.4 EMERGENCY COOLING POND

LIMITING CONDITION FOR OPERATION

3.7.4.1 The emergency cooling pond (ECP) shall be OPERABLE[1] with:

a. A minimum contained water volume of 70 acre-feet.

b. An average water temperature of ≤ 100 °F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the volume and/or temperature requirements of the above specification not satisfied or, with the requirements of Action b not met, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. If degradation is noted pursuant to 4.7.4.1.d below or by other inspection, perform an evaluation to determine that the ECP remains acceptable for continued operation within 7 days.

SURVEILLANCE REQUIREMENTS

4.7.4.1 The ECP shall be determined OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the indicated water level of the ECP is greater than or equal to that required for an ECP volume of 70 acre-feet.

b. In accordance with the Surveillance Frequency Control Program during the period of June 1 through September 30 by verifying that the pond's average water temperature at the point of discharge from the pond is within its limit.

c. In accordance with the Surveillance Frequency Control Program by making soundings of the pond and verifying:

1. A contained water volume of ECP ≥ 70 acre-feet, and

2. The minimum indicated water level needed to ensure a volume of 70 acre-feet is maintained.

d. In accordance with the Surveillance Frequency Control Program by performance of a visual inspection of the ECP to verify conformance with design requirements.

Note 1: The ECP may be considered OPERABLE on a one-time basis for up to 65 days during upgrade of the ECP supply piping to the SWS intake bays provided:

a. A loss of Lake Dardanelle event is not in progress, and

b. A temporary pumping system is capable of supplying the SWS from the ECP. The temporary pumping system may be unavailable for testing or necessary maintenance provided its availability is restored within 72 hours, and

SURVEILLANCE REQUIREMENTS (Continued)

c. The compensatory measures described in the ANO correspondence letter 0CAN022201, dated February 17, 2022, Enclosure, Attachment 4 shall be implemented. Failure to meet one or more of the continuing compliance compensatory measures is acceptable provided the measure(s) is/are restored within 72 hours.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent control room emergency ventilation and air conditioning systems shall be OPERABLE. (Note 1)

APPLICABILITY: MODES 1, 2, 3, 4, or during movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4

a. With one control room emergency air conditioning system (CREACS) inoperable, restore the inoperable system to OPERABLE status within 30 days.

b. With one control room emergency ventilation system (CREVS) inoperable for reasons other than ACTION d, restore the inoperable system to OPERABLE status within 7 days.

c. With one CREVS inoperable for reasons other than ACTION d and one CREACS inoperable, restore the inoperable CREVS to OPERABLE status within 7 days and restore the inoperable CREACS to OPERABLE status within 30 days.

d. With one or more CREVS inoperable due to an inoperable CRE boundary:

   1. Immediately initiate action to implement mitigating actions, and

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

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

e. With two CREVS inoperable for reasons other than ACTION d (Note 2):

   1. Immediately initiate action to implement mitigating actions, and

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

   3. Within 24 hours, restore at least one CREVS to OPERABLE status.

f. With two CREACS inoperable (Note 2), restore at least one CREACS to OPERABLE status within 24 hours.

With ACTIONS a, b, c, d, e, and/or f not met, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Note 1: The control room envelope (CRE) boundary may be open intermittently under administrative controls.

Note 2: ACTION e is not applicable if the second CREVS is intentionally made inoperable. ACTION f is not applicable if the second CREACS is intentionally made inoperable.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

During Movement of Irradiated Fuel Assemblies or Movement of New Fuel Assemblies over Irradiated Fuel Assemblies

g. With one CREACS inoperable, restore the inoperable system to OPERABLE status within 30 days or immediately place the OPERABLE system in operation; otherwise, suspend all activities involving the movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

h. With one CREVS inoperable, restore the inoperable system to OPERABLE status within 7 days or immediately place the control room in the emergency recirc mode of operation; otherwise, suspend all activities involving the movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

i. With one CREVS inoperable for reasons other than ACTION d and one CREACS inoperable:

   1. restore the inoperable CREVS to OPERABLE status within 7 days or immediately place the CRE in the emergency recirc mode of operation, and

   2. restore the inoperable CREACS to OPERABLE status within 30 days or immediately place the OPERABLE system in operation;

   3. otherwise, suspend all activities involving the movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

j. With both CREACS inoperable, immediately suspend all activities involving the movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

k. With both CREVS inoperable or with one or more CREVS inoperable due to an inoperable CRE boundary, immediately suspend all activities involving the movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 Each control room emergency air conditioning system shall be demonstrated OPERABLE:

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

1. Starting each unit from the control room, and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤ 84 °F D.B.

b. In accordance with the Surveillance Frequency Control Program by verifying a system flow rate of 9900 cfm ± 10%.

4.7.6.1.2 Each control room emergency air filtration system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the system operates for at least 15 minutes.

b. In accordance with the Surveillance Frequency Control Program by verifying that on a control room high radiation signal, either actual or simulated, the system automatically isolates the control room and switches into a recirculation mode of operation, except for dampers and valves that are locked, sealed, or otherwise secured in the actuated position.

c. By performing the required Control Room Emergency Ventilation filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

d. Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

3/4.8.1 A.C. SOURCES

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system and

b. Two separate and independent diesel generators each with:

1. A day fuel tank containing a minimum volume of 300 gallons of fuel,

2. A separate fuel storage system, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

NOTE: Specification 3.0.4.b is not applicable to diesel generators.

a. With one offsite A.C. circuit of the above required A.C. electrical power sources inoperable, perform the following:

1. Demonstrate the OPERABILITY of the remaining offsite A.C. circuit by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and

2. Within 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required features(s), declare required features(s) with no offsite power available inoperable when its redundant required features(s) is inoperable, and

3. Restore the offsite A.C. circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. Startup Transformer No. 2 may be removed from service for up to 30 days as part of a preplanned preventative maintenance schedule. The 30-day allowance may be applied not more than once in a 10-year period.

3/4.8.1 A.C. SOURCES

LIMITING CONDITION FOR OPERATION

b. With one diesel generator of the above required A.C. electrical power source inoperable, perform the following:

1. Demonstrate the OPERABILITY of both the offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and

2. Within 4 hours from discovery of one required diesel generator inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) supported by the inoperable diesel generator inoperable when its redundant required feature(s) is inoperable, and

3. Demonstrate the OPERABILITY of the remaining OPERABLE diesel generator within 24 hours by:

   i. Determining the OPERABLE diesel generator is not inoperable due to a common cause failure, or

   ii. Perform Surveillance Requirement 4.8.1.1.2.a.4 unless:

      a. The remaining diesel generator is currently in operation, or

      b. The remaining diesel generator has been demonstrated OPERABLE within the previous 24 hours, and

4. Restore the diesel generator to OPERABLE status within 14 days (See Note 1) or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Note 1 – If the Alternate A.C. Diesel Generator (AACDG) is determined to be inoperable during this period, then a 72 hour restoration or Risk Informed Completion Time period is applicable until either the AACDG or the diesel generator is returned to operable status (not to exceed 14 days or the Risk Informed Completion Time from the initial diesel generator inoperability).

c. With one offsite A.C. circuit and one diesel generator of the above required A.C. electrical power sources inoperable (see Note 2), perform the following:

1. Demonstrate the OPERABILITY of the remaining offsite A.C. circuit by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and,

2. Within 4 hours from discovery of one required diesel generator inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) supported by the inoperable diesel generator inoperable if its redundant required feature(s) is inoperable, and

3. If the diesel generator became inoperable due to any cause other than preplanned preventative maintenance or testing, then

   i. Demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours, except when:

      a. The remaining diesel generator is currently in operation, or

      b. The remaining diesel generator has been demonstrated OPERABLE within the previous 8 hours, and

4. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or in accordance with the Risk Informed Completion Time Program, and

5. Restore the remaining inoperable A.C. Source to an OPERABLE status (Offsite A.C. Circuit within 72 hours or in accordance with the Risk Informed Completion Time Program, or Diesel Generator within 14 days or in accordance with the Risk Informed Completion Time Program (see b.4, Note 1)), based on the time of the initiating event that caused the inoperability.

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

Note 2 – Enter applicable ACTIONs of LCO 3.8.2.1, "A.C. Distribution – Operating," when ACTION c is entered with no AC power to any train.

LIMITING CONDITION FOR OPERATION

d. With two offsite A.C. circuits of the above required A.C. electrical power sources inoperable, perform the following:

   1. Perform Surveillance Requirement 4.8.1.1.2.a.4 on the diesel generators within the next 8 hours except when:

      i. The diesel generators are currently in operation, or

      ii. The diesel generators have been demonstrated OPERABLE within the previous 8 hours, and

   2. Within 12 hours from discovery of two required offsite A.C. circuits inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) inoperable when its redundant required feature(s) is inoperable, and

   3. Restore one of the inoperable offsite A.C. circuits to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, and

   4. Restore both A.C. circuits within 72 hours or in accordance with the Risk Informed Completion Time Program of the initiating event,

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

e. With two diesel generators of the above required A.C. electrical power sources inoperable, perform the following:

   1. Demonstrate the OPERABILITY of the two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and

   2. Restore one of the inoperable diesel generators to OPERABLE status within 2 hours, and

   3. Restore the remaining inoperable diesel generator within 14 days or in accordance with the Risk Informed Completion Time Program (see b.4, Note 1) of the initiating event.

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

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

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

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

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE: (Note 1)

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

1. Verifying the fuel level in the day fuel tank.

2. deleted

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from a standby condition and accelerates to at least 900 rpm in ≤ 15 seconds. (Note 2)

5. Verifying the generator is synchronized, loaded to an indicated 2600 to 2850 Kw and operates for ≥ 60 minutes. (Notes 3 & 4)

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

b. deleted

Note 1

All planned diesel generator starts for the purposes of these surveillances may be preceded by prelube procedures.

Note 2

This diesel generator start from a standby condition in ≤ 15 sec. shall be accomplished at least once every 184 days. All other diesel generator starts for this surveillance may be in accordance with vendor recommendations.

Note 3

Diesel generator loading may be accomplished in accordance with vendor recommendations such as gradual loading.

Note 4

Momentary transients outside this load band due to changing loads will not invalidate the test. Load ranges are allowed to preclude over- loading the diesel generators.

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

1. Deleted

2. Verifying during shutdown that the automatic sequence time delay relays are OPERABLE at their setpoint ± 10% of the elapsed time for each load block.

3. Verifying during shutdown the generator capability to reject a load of greater than or equal to its associated single largest post-accident load, and maintain voltage at 4160 ± 500 volts and frequency at 60 ± 3 Hz.

4. Verifying during shutdown the generator capability to reject a load of 2850 Kw without exceeding 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is lower.

5. Simulating during shutdown 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 from a standby condition on the undervoltage auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected shutdown loads through the time delay relays and operates for ≥ 5 minutes while its generator is loaded with the shutdown loads.

6. Verifying during shutdown that on a Safety Injection Actuation Signal (SIAS) actuation test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for ≥ 5 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

7. Verifying during shutdown that all diesel generator trips, except engine overspeed, lube oil pressure, generator differential, and engine failure to start, are automatically bypassed upon a Safety Injection Actuation Signal.

8. Simulating during shutdown a loss of offsite power in conjunction with SIAS and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts from a standby condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency (accident) loads through the Time Delay Relays and operates for ≥ 5 minutes while its generator is loaded with the emergency loads.

9. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to an indicated 3000 to 3200 Kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 2600 to 2850 Kw (Notes 3 & 4). Within 5 minutes after completing this 24 hour test, perform 4.8.1.1.2.a.4. (Note 5)

10. Verifying that the auto-connected loads to each diesel generator do not exceed the 2 hour rating of 3135 Kw.

Note 3
Diesel generator loading may be accomplished in accordance with vendor recommendations, such as gradual loading.

Note 4
Momentary transients outside this load band due to changing loads will not invalidate the test. Load ranges are allowed to preclude overloading the diesel generators.

Note 5
If this test is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test, instead, the diesel generator may be operated at 2600 to 2850 Kw until internal temperatures stabilize but not less than 2 hours, then perform test 4.8.1.1.2.a.4 within 5 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

11. Verifying during shutdown the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source, and

    c) Proceed through its shutdown sequence.

12. Verifying during shutdown 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 auto-connected emergency (accident) loads with offsite power.

13. 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.

d. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 900 rpm in $\leq$ 15 seconds.

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 300 gallons of fuel,

2. A fuel storage system, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6, or during movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies.

ACTION:

Note: Enter applicable ACTIONs of LCO 3.8.2.2, "A.C. Distribution – Shutdown," and LCO 3.8.2.4, "D.C. Sources – Shutdown," with one required train de-energized.

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

SURVEILLANCE REQUIREMENT

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for Requirement 4.8.1.1.2a.5.

LIMITING CONDITION FOR OPERATION

3.8.1.3 The stored diesel fuel oil shall be within limits for each required diesel generator.

APPLICABILITY: When associated diesel generator is required to be OPERABLE.

ACTION:

With the volume of the stored diesel fuel oil less than 22,500 gallons for either fuel oil storage tank or the new or stored fuel oil properties outside the limits of the Diesel Fuel Oil Testing Program, perform the following as appropriate: (Note – Separate ACTION entry is allowed for each diesel generator.)

1. If one or more fuel storage tanks contain less than 22,500 gallons and greater than 17,446 gallons, restore the fuel oil volume to within limits within 48 hours.

2. If the stored fuel oil total particulates are not within limits for one or more diesel generators, restore fuel oil total particulates to within limits within 7 days.

3. If new fuel oil properties are not within limits for the one or more diesel generators, restore stored fuel oil properties to within limits within 30 days.

4. If ACTION 1 is not met within the allowable outage time or is outside the allowable limits, or if ACTION 2 or 3 is not met within the allowable outage time, then immediately declare the associated diesel generator inoperable.

SURVEILLANCE REQUIREMENTS

4.8.1.3.1 In accordance with the Surveillance Frequency Control Program verify the fuel oil storage tank contains ≥ 22,500 gallons of fuel.

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

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION – OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open between redundant busses:

4160 volt Emergency Bus # 2A3

4160 volt Emergency Bus # 2A4

480 volt Emergency Bus # 2B5

480 volt Emergency Bus # 2B6

120 volt A.C. Vital Bus # 2RS1

120 volt A.C. Vital Bus # 2RS2

120 volt A.C. Vital Bus # 2RS3

120 volt A.C. Vital Bus # 2RS4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Note: Enter applicable ACTIONs of LCO 3.8.2.3, "DC Sources – Operating" for DC train(s) made inoperable by inoperable power distribution subsystems.

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

A.C. DISTRIBUTION – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE:

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Load Center Bus

4 - 480 volt Motor Control Center Busses

2 - 120 volt A.C. Vital Busses

APPLICABILITY: MODES 5 and 6, or during movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, immediately declare affected required features inoperable OR:

a. Immediately suspend the movement of recently irradiated fuel assemblies, the movement of new fuel assemblies over recently irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM or boron concentration, and

b. Immediately initiate actions to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status, and

c. Immediately declare associated required shutdown cooling subsystem(s) inoperable and not in operation.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

DC SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required full capacity chargers inoperable:

   i. Restore the battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, and

   ii. Verify battery float current ≤ 2 amps once per 12 hours.

b. With one DC electrical power subsystem inoperable for reasons other than ACTION ‘a’ above, restore the inoperable DC electrical power subsystem to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program.

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

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 In accordance with the Surveillance Frequency Control Program by verifying that the battery terminal voltage is greater than or equal to the minimum established float voltage.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.2.3.2 In accordance with the Surveillance Frequency Control Program by verifying that each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours or, by verifying that each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.

4.8.2.3.3 In accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is adequate to supply, and maintain in OPERABLE status, required emergency loads for the design duty cycle when subjected to a battery service test. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this Surveillance. The battery performance discharge test required by Surveillance Requirement 4.8.3.6 may be performed in lieu of the battery service test once per 60 months.

DC SOURCES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following DC electrical equipment and bus shall be energized and OPERABLE:

1 - 125-volt DC bus, and

1 - 125-volt battery bank and charger supplying the above DC bus.

APPLICABILITY: MODES 5 and 6, or during movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies.

ACTION:

a. With the required battery charger inoperable:

   i. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, and

   ii. Verify battery float current ≤ 2 amps once per 12 hours.

b. With the requirements of ACTION ‘a’ not met or with the above complement of DC equipment and bus otherwise inoperable, immediately declare affected required features inoperable OR:

   i. Immediately suspend the movement of recently irradiated fuel assemblies, the movement of new fuel assemblies over recently irradiated fuel assemblies, and any operations involving positive reactivity additions that could result in loss of required SDM or boron concentration, and

   ii. Immediately initiate actions to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status, and

   iii. Immediately declare associated required shutdown cooling subsystem(s) inoperable and not in operation.

SURVEILLANCE REQUIREMENTS

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

4.8.2.4.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirements 4.8.2.3.1, 4.8.2.3.2, and 4.8.2.3.3; however, while each of these Surveillance Requirements must be met, Surveillance Requirements 4.8.2.3.2 and 4.8.2.3.3 are not required to be performed.

ELECTRICAL POWER SYSTEMS

BATTERY PARAMETERS

LIMITING CONDITION FOR OPERATION

3.8.3 Battery parameters for the Train A and Train B electrical power subsystem batteries shall be within the limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTION:

a. With one battery with one or more battery cells float voltage < 2.07 V:

   i. Within 2 hours perform Surveillance Requirements 4.8.2.3.1 and 4.8.3.1, and

   ii. Within 24 hours restore affected cell voltage to ≥ 2.07 V.

b. With one battery with float current > 2 amps:

   i. Within 2 hours perform Surveillance Requirement 4.8.2.3.1, and

   ii. Within 12 hours restore battery float current to ≤ 2 amps.

c. With one battery with one or more cells electrolyte level less than minimum established design limits:

   i. Within 8 hours restore electrolyte level to above top of plates[1], and

   ii. Within 12 hours verify no evidence of leakage[1], and

   iii. Within 31 days restore electrolyte level to greater than or equal to minimum established design limits.

d. With one battery with pilot cell electrolyte temperature less than minimum established design limits, restore battery pilot cell electrolyte temperature to greater than or equal to minimum established design limits within 12 hours.

e. With both batteries with battery parameters not within limits, restore battery parameters for at least one battery to within limits within 2 hours.

f. With the requirements of ACTION 'a', 'b', 'c', 'd', or 'e' not met, or with one battery with one or more battery cells float voltage < 2.07 V and float current > 2 amps, immediately declare the battery inoperable.

Note 1: Only required if electrolyte level is below the top of the plates. If electrolyte level is below the top of the plates, ACTION c.ii shall be performed.

SURVEILLANCE REQUIREMENTS

4.8.3.1 In accordance with the Surveillance Frequency Control Program by verifying that each battery float current is ≤ 2 amps. This Surveillance is not required when battery terminal voltage is less than the minimum established float voltage of Surveillance Requirement 4.8.2.3.1.

4.8.3.2 In accordance with the Surveillance Frequency Control Program by verifying that each battery pilot cell float voltage is ≥ 2.07 V.

4.8.3.3 In accordance with the Surveillance Frequency Control Program by verifying that each battery connected cell electrolyte level is greater than or equal to minimum established design limits.

4.8.3.4 In accordance with the Surveillance Frequency Control Program by verifying that each battery pilot cell temperature is greater than or equal to minimum established design limits.

4.8.3.5 In accordance with the Surveillance Frequency Control Program by verifying that each battery connected cell float voltage is ≥ 2.07 V.

4.8.3.6 In accordance with the Surveillance Frequency Control Program by verifying the battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this Surveillance. In addition, the performance discharge test shall be performed:

a. At least once per 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating, and

b. At least once per 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of the reactor coolant system and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of ≥ 2500 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving positive reactivity changes and initiate and continue boration at ≥ 40 gpm of ≥ 2500 ppm boric acid solution until K_{eff} is reduced to ≤ 0.95 or the boron concentration is restored to ≥ 2500 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

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

4.9.1.2 The boron concentration of the reactor coolant and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

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* Only applicable to the refueling canal when connected to the RCS.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, 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 or more of the above required monitors inoperable, immediately suspend positive reactivity additions.

   AND

   Suspend movement of fuel, sources, and reactivity control components within the reactor vessel.[1]

b. With both of the above required monitors inoperable, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,

b. A CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program, and

c. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of the movement of recently irradiated fuel assemblies or the movement of new fuel assemblies over recently irradiated fuel assemblies.

Note 1: Fuel assemblies, sources, and reactivity control components may be moved if necessary to restore an inoperable source range neutron flux monitor or to complete movement of a component to a safe condition.

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REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door is capable* of being closed,

b. A minimum of one door in each airlock is capable* of being closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed* by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable* of being closed by an OPERABLE containment purge and exhaust isolation system.

APPLICABILITY: During movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies within the Containment Building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies in the Containment Building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required conditions within 72 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies in the Containment Building.

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* Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. Administrative controls shall ensure that appropriate personnel are aware that when containment penetrations, including both personnel airlock doors and/or the equipment door are open, a specific individual(s) is designated and available to close the penetration following a required evacuation of containment, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door and/or the equipment door be capable of being quickly removed.

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REFUELING OPERATIONS

SHUTDOWN COOLING AND COOLANT CIRCULATION

SHUTDOWN COOLING – ONE LOOP

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one shutdown cooling loop in operation, except as provided in b. below, suspend operations involving an increase in the reactor decay heat load or that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period provided no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 2000 gpm in accordance with the Surveillance Frequency Control Program.

REFUELING OPERATIONS

SHUTDOWN COOLING – TWO LOOPS

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet.

ACTION:

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required shutdown cooling loops shall be determined OPERABLE per the INSERVICE TESTING PROGRAM.

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* The normal or emergency power source may be inoperable for each shutdown cooling loop.

REFUELING OPERATIONS

WATER LEVEL – REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the elevation corresponding to the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies within the Containment Building.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the Containment Building. The provisions of Specification 3.0.3 are not applicable.

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 in accordance with the Surveillance Frequency Control Program thereafter during movement of fuel assemblies within the Containment Building.

REFUELING OPERATIONS

SPENT FUEL POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.10 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 requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pool areas and restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the spent fuel pool.

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FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12.a Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.95 w/o U-235. The provisions of Specification 3.0.3 are not applicable.

3.9.12.b Storage in the spent fuel pool shall be further restricted by the limits specified in Table 3.9-1. The provisions of Specification 3.0.3 are not applicable.

3.9.12.c The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 2000 parts per million.

APPLICABILITY: During storage of fuel in the spent fuel pool

ACTION:

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined a fuel assembly has been placed in an incorrect location until such time as the correct storage location is determined. Move the assembly to its correct location before resumption of any other fuel movement.

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than 2001 ppm, until such time as the boron concentration is increased to 2001 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.12.a Verify all fuel assemblies to be placed in the spent fuel pool have an initial enrichment of less than or equal to 4.95 w/o U-235 by checking the assemblies' design documentation.

4.9.12.b Verify all fuel assemblies to be placed in the spent fuel pool are within the limits of Table 3.9-1 by checking the assemblies' design and burnup documentation.

4.9.12.c Verify in accordance with the Surveillance Frequency Control Program the spent fuel pool boron concentration is greater than 2000 ppm.

4.9.12.d Verify Metamic properties are in accordance with, and are maintained within the limits of, the Metamic Coupon Sampling Program.

Table 3.9-1
SFP Loading Restrictions

Region 1
No loading restrictions other than U-235 enrichment.

Region 2
Minimum Burnup at Varying Initial U-235 Enrichment and Cooling Time (Notes 1 and 2)

Enrichment (Wt% U-235)	2.0	2.5	3.0	3.5	4.0	4.5	4.95
Cooling Time (Years)	Minimum Burnup (GWD/MTU)						
0	6.4	13.7	21.2	27.9	33.8	40.7	46.8
1	NC	NC	NC	27.1	33.0	39.5	45.8
2	NC	NC	NC	26.7	32.5	38.9	44.8
3	NC	NC	NC	26.5	32.1	38.4	44.3
4	NC	NC	NC	26.2	31.6	38.0	43.7
5	5.9	12.6	19.3	26.1	31.2	37.4	43.1
10	5.7	12.0	18.4	25.3	29.7	35.6	41.1
15	5.6	11.6	18.1	25.0	29.1	34.4	39.7
20	5.4	11.4	17.5	24.3	28.6	34.0	38.9

Region 2
Minimum Burnup versus U-235 Enrichment for Peripheral Cells
with Spent Fuel at 0 Years Cooling Time (Note 3)

Enrichment (Wt% U-235)	2.0	2.5	3.0	3.5	4.0	4.5	4.95
Minimum Burnup (GWD/MTU)	0	4.7	9.7	15.0	21.8	27.6	33.3

Rack Interface Allowances

1. Region 1 to Region 2, fresh fuel checkerboard in Region 2 is allowed. Spent fuel in Region 2 is allowed.
2. Region 2 Racks – a fresh fuel checkerboard and uniform spent fuel loading may be placed adjacent to each other in the same rack. If both patterns are placed in a single rack, no fresh fuel assembly may be placed with more than one face adjacent to a spent fuel assembly.
3. Region 2 Racks – if adjacent racks contain a checkerboard of fresh fuel assemblies, the checkerboard must be maintained across the gap, i.e., fresh fuel assemblies may not face each other across a gap.
4. Region 2 Racks – one rack may contain a checkerboard of fresh fuel and empty storage locations and the adjacent rack may contain spent fuel with no loading restrictions.

Notes: 1. Linear interpolation between burnups for a given cooling time is allowed. However, linear interpolation between cooling times is not allowed, therefore the cooling time of a given assembly must be rounded down to the nearest cooling time.

2. NC = Not Calculated, if any fuel assembly is within these limits, use 0 cooling time and interpolate for enrichments to determine loading restrictions per note 1.

3. Linear interpolation between burnups is allowed.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each CEA required either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 14 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The linear heat rate limit shall be maintained by either:

   1. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

   2. Operating within the region of acceptable operation as specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service.)

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With any of the above limits being exceeded while any of the above requirements are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of the above Specification, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which any of the above requirements are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within its limits during PHYSICS TESTS above 5% of RATED THERMAL POWER in which any of the above requirements are suspended.

SPECIAL TEST EXCEPTIONS

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at ≤ 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during startup and PHYSICS TESTS.

4.10.3.2 Each wide range logarithmic and power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the incore detection system during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

SPECIAL TEST EXCEPTIONS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.5 The minimum temperature for criticality limits of Specification 3.1.1.5 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Linear Power Level - High neutron flux monitoring channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation shown on Figure 3.4-2.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 in accordance with the Surveillance Frequency Control Program.

4.10.5.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program.

4.10.5.3 Each Logarithmic Power Level and Linear Power Level channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 Site Location

The site for Arkansas Nuclear One is located in Pope County, Arkansas on the north bank of the Dardanelle Reservoir (Arkansas River), approximately 6 miles west-northwest of Russellville, AR. The exclusion area boundary shall have a radius of 0.65 statute miles from the center of the reactor.

5.2 Reactor Core

5.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO^{TM} or Optimized ZIRLO^{TM} clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. Other cladding material may be used with an approved exemption.

5.2.2 Control Element Assemblies

The reactor core shall contain 81 control element assemblies. The control material shall be boron carbide and silver-indium-cadmium as approved by the NRC.

5.3 Fuel Storage

5.3.1 Spent Fuel Storage Rack Criticality

The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies stored in the spent fuel pool in accordance with Specification 3.9.12;

b. $k_{eff} \leq 0.95$ if fully flooded with 452 ppm borated water, which includes an allowance for uncertainties as described in Section 9.1 of the SAR; and

c. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the SAR; and

d. A nominal 9.8 inch center to center distance between fuel assemblies placed in the storage racks.

5.3.2 New Fuel Storage Rack Criticality

The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.95 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the SAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the SAR; and

d. A nominal 26 inch center to center distance between fuel assemblies placed in the storage racks.

5.3.3 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 399' 10½".

5.3.4 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 988 fuel assemblies.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is in MODE 1, 2, 3, or 4. With the unit not in MODES 1, 2, 3, or 4, an individual with an active SRO or Reactor Operator license shall be designated as responsible for the control room command function.

6.2 ORGANIZATION

6.2.1 ONSITE AND OFFSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power unit.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the unit specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Safety Analysis Report (SAR);

b. The Plant Manager Operations shall be responsible for overall safe operation of the unit and shall have control over those onsite activities necessary for safe operation and maintenance of the unit;

c. A specified corporate executive shall have corporate responsibility for overall unit nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the unit to ensure nuclear safety. The specified corporate executive shall be identified in the SAR; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

a. A non-licensed operator shall be on site when fuel is in the reactor and two additional non-licensed operators shall be on site when the reactor is in MODES 1, 2, 3, or 4.

b. The minimum shift crew composition for licensed operators shall meet the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) for one unit, one control room.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) for one unit, one control room, and 6.2.2.a and 6.2.2.f 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.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. The operations manager or the assistant operations manager shall hold a SRO license.

f. When in MODES 1, 2, 3, or 4 an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operations of the unit. This individual shall meet the qualifications specified by ANSI/ANS 3.1-1993 as endorsed by RG 1.8, Rev. 3, 2000.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions with exceptions specified in the Entergy Quality Assurance Program Manual (QAPM).

6.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.4 PROCEDURES

6.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Section 7.1 of Generic Letter 82-33;

c. Deleted

d. All programs specified in Specification 6.5; and

e. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with the CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

6.5 PROGRAMS AND MANUALS

The following programs shall be established, implemented, and maintained.

6.5.1 Offsite Dose Calculation Manual (ODCM)

The 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 and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

    2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the ANO general manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall indicate the date (i.e., month and year) the change was implemented.

6.5 PROGRAMS AND MANUALS

6.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The program shall include the following:

c. Preventive maintenance and periodic visual inspection requirements; and

d. Integrated leak test requirements for each system at a Frequency in accordance with the Surveillance Frequency Control Program. The provisions of Surveillance Requirements 4.0.2 are applicable.

6.5.3 Iodine Monitoring

This program provides controls that ensure the capability to accurately determine the airborne iodine concentration under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for monitoring; and

c. Provisions for maintenance of sampling and analysis equipment.

6.5.4 Radioactive Effluent Controls Program

This program conforms 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 shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS, conforming to 10 CFR 50, Appendix I;

6.5.4 Radioactive Effluent Controls Program (continued)

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.5.5 Component Cyclic or Transient Limit Program

This program provides controls to track the SAR Section 5.2.1.5, cyclic and transient occurrences to ensure that components are maintained within the design limits.

6.5.6 Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in prestressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include the use of baseline measurements from initial operation. The Containment Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with the ASME Code, Section XI, Subsection IWL and 10 CFR 50.55a.

The provisions of SR 4.0.3 are applicable to the Containment Tendon Surveillance Program inspection frequencies.

6.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. The volumetric examination per Regulatory Position C.4.b.1 will be performed on approximately 10-year intervals.

6.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Decay Tanks, the quantity of radioactivity contained in the Waste Gas Decay Tanks, and the quantity of radioactivity contained in unprotected temporary outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology contained in Revision 2 of NUREG-0800, "Standard Review Plan," Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure," Revision 0, July 1981. The liquid radwaste quantities shall be determined in accordance with the ODCM.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Decay Tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion),

b. A surveillance program to ensure that the quantity of radioactivity contained in each Waste Gas Decay Tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents, and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

6.5.9 Steam Generator (SG) Program

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm through any one SG.

   3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

6.5.9 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary leakage.

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6.5.10 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedure for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

6.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safeguards (ES) ventilation systems filters at the frequencies specified in Regulatory Guide 1.52, Revision 2. The VFTP is applicable to the Control Room Emergency Ventilation System (CREVS).

a. Demonstrate that an inplace cold DOP test of the high efficiency particulate (HEPA) filters shows ≥ 99.95% DOP removal when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flowrate of 2000 cfm ± 10%.

b. Demonstrate that an inplace halogenated hydrocarbon test of the charcoal adsorbers shows ≥ 99.95% halogenated hydrocarbon removal when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system design flow rate of 2000 cfm ± 10%.

c. Demonstrate that a laboratory test of a sample of the charcoal adsorber meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30 °C and 95% relative humidity for a methyl iodide penetration of:

   1. ≤ 2.5% for 2 inch charcoal adsorber beds, and

   2. ≤ 0.5% for 4 inch charcoal adsorber beds,

   when obtained as described in Regulatory Guide 1.52, Revision 2.

d. Demonstrate that the pressure drop across the combined HEPA filters, other filters in the system, and charcoal adsorbers is < 6 inches of water when tested at the following system design flowrate of 2000 cfm ± 10%.

The provision of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

6.5.12 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem Total Effective Dose Equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of Specification 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

6.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

    1. an API gravity or an absolute specific gravity within limits,

    2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

    3. water and sediment within limits;

b. Within 31 days following addition of new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested based on ASTM D-2276, Method A-2 or A-3 at a Frequency in accordance with the Surveillance Frequency Control Program; and

d. The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies.

6.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

    1. A change in the TS incorporated in the license or

    2. A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the SAR.

d. Proposed changes that do not meet the criteria of 6.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.5.15 Battery Monitoring and Maintenance Program

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision 2 (RG), with RG exceptions and program provisions as identified below:

a. The program allows the following RG 1.129, Revision 2 exceptions:

1. Battery temperature correction may be performed before or after conducting discharge tests.

2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.

3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery."

4. In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."

5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration", the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < 2.13 V;

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cell has been found to be < 2.13 V;

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

6.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008. The next Type A test performed after the November 30, 2000 Type A test shall be performed no later than November 30, 2015.

In addition, the containment purge supply and exhaust isolation valves shall be leakage rate tested prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 58 psig.

The maximum allowable containment leakage rate, L_a, shall be 0.1% of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. Leakage rate for each Personnel Air Lock door is $\leq 0.01\ L_a$ when pressurized to $\geq$ 10 psig.

   3. A seal contact check for each Emergency Escape Air Lock door, consisting of a verification of continuous contact between the seals and the sealing surfaces.

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.

6.5.17 Metamic Coupon Sampling Program

A coupon surveillance program will be implemented to maintain surveillance of the Metamic absorber material under the radiation, chemical, and thermal environment of the SFP. The purpose of the program is to establish the following:

- Coupons will be examined on a two year basis for the first three intervals with the first coupon retrieved for inspection being on or before October 31, 2009 and thereafter at increasing intervals over the service life of the inserts.

- Measurements to be performed at each inspection will be as follows:

  a. Physical observations of the surface appearance to detect pitting, swelling or other degradation,

  b. Length, width, and thickness measurements to monitor for bulging and swelling

  c. Weight and density to monitor for material loss, and

  d. Neutron attenuation to confirm the B-10 concentration or destructive chemical testing to determine the boron content.

- The provisions of SR 4.0.2 are applicable to the Metamic Coupon Sampling Program.

- The provisions of SR 4.0.3 are not applicable to the Metamic Coupon Sampling Program.

6.5.18 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

6.5.19 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system ACTIONs. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

c. Provisions to ensure that an inoperable supported system's allowed outage time is not inappropriately extended as a result of multiple support system inoperabilities, and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate ACTIONs of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ACTIONs to enter are those of the support system.

6.5.20 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1 and 2;

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

   1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

   2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

   3. Revising the RICT is not required If the plant configuration change would lower plant risk and would result in a longer RICT.

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

   1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

   2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods approved for use with this program, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

6.6 REPORTING REQUIREMENTS

6.6.1 DELETED

6.6.2 Annual Radiological Environmental Operating Report

(Note: A single submittal may be made for ANO. The submittal should combine sections common to both units.)

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

6.6.3 Radioactive Effluent Release Report

(Note: A single submittal may be made for ANO. The submittal shall combine sections common to both units. The submittal shall specify the releases of radioactive material from each unit.)

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

6.6.4 Post Accident Monitoring Report

When a report is required by TS Table 3.3-10, "Post-Accident Monitoring Instrumentation," Action 1 or Action 3, a report shall be submitted within the following 14 days. 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 of the Function to OPERABLE status.

6.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle, and shall be documented in the COLR for the following:

3.1.1.1 Shutdown Margin – T_{avg} > 200°F
3.1.1.2 Shutdown Margin - $T_{avg} \leq$ 200°F
3.1.1.4 Moderator Temperature Coefficient
3.1.3.1 CEA Position
3.1.3.6 Regulating and Group P CEA Insertion Limits
3.2.1 Linear Heat Rate
3.2.3 Azimuthal Power – Tq
3.2.4 DNBR Margin
3.2.7 Axial Shape Index

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1) "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" (WCAP-11596-P-A), "ANC: A Westinghouse Advanced Nodal Computer Code" (WCAP-10965-P-A), and "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery" (WCAP-10965-P-A Addendum 1) (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.4 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, and 3.2.4.b for DNBR Margin).

2) "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A (Methodology for Specification 3.1.3.6 for Regulating and Group P CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

3) "Modified Statistical Combination of Uncertainties, CEN-356(V)-P-A, Revision 01-P-A (Methodology for Specification 3.2.4.c and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

4) "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

5) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6.6.5 CORE OPERATING LIMITS REPORT (COLR) (Continued)

6) "Technical Manual for the CENTS Code," WCAP-15996-P-A, Rev. 1 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

7) "Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A (modifies CENPD-132-P and CENPD-137-P as methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

8) "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (may be used as a replacement for the PHOENIX-P lattice code as the methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.4 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, and 3.2.4.b for DNBR Margin).

9) "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," WCAP-16072-P-A (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Tilt, and 3.2.7 for ASI).

10) "CE 16 x 16 Next Generation Fuel Core Reference Report," WCAP-16500-P-A (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, 3.2.4.b, 3.2.4.c and 3.2.4.d for DNBR Margin, and 3.2.7 for ASI).

11) "Optimized ZIRLO™," WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

12) "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," WCAP-16523-P-A (Methodology for Specification 3.2.4.b, 3.2.4.c and 3.2.4.d for DNBR Margin).

13) "ABB Critical Heat Flux Correlations for PWR Fuel," CENPD-387-P-A (Methodology for Specification 3.2.4.b, 3.2.4.c and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

c. The core operating limits shall be determined such that all applicable limits (e.g. fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.6.6 Containment Inspection Report

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Tendon Surveillance 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 applicability of the conditions to the other unit, 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 date of necessary repairs, and the extent, nature, and frequency of additional examinations.

6.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with the Specification 6.5.9, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

   1. The nondestructive examination techniques utilized;

   2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

   3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

   4. The number of tubes plugged during the inspection outage.

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG; and

f. The results of any SG secondary side inspections.

6.6.8 Specific Activity

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded the results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

6.7 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP), or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

   2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

   3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

6.7 HIGH RADIATION AREA (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.7 HIGH RADIATION AREA (continued)

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP, or equivalent, while in the area by means of closed circuit television, or personnel qualified in radiation protection procedures responsible for controlling personnel radiation exposure in the area and with the means to communicate with individuals in the area who are covered by such surveillance.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

Issued with original License.

ATTACHMENT 2 TO LICENSE NPF-6

Preoperational Tests, Startup Tests, and Other Items Which Must be Completed Prior to Loading Fuel

Authorized to proceed to Mode 6 and 5

This attachment identifies certain preoperational tests, startup tests, and other items which must be completed to the Commission's satisfaction prior to proceeding to Operational Mode 6 (Fuel Loading). Arkansas Power & Light Company shall not proceed into Operational Mode 6 without prior written authorization from the Commission.

See authorization Tab for further info

1. Completion of installation of excore nuclear instrumentation.

2. Completion of preoperational test 2.600.12A, "Special Tests of Boration/Dilution System."

3. Resolution of questions on installation of fire barriers at the ends of conduit rather than at wall penetrations.

ATTACHMENT 2 TO LICENSE NPF-6

Preoperational Tests, Startup Tests, and Other Items Which Must be Completed Prior to Proceeding To Succeeding Operational Modes

This attachment identifies certain preoperational tests, startup tests, and other items which must be completed to the Commission's satisfaction prior to proceeding to certain specified Operational Modes. Arkansas Power & Light Company shall not proceed beyond the authorized Operational Modes without prior written authorization from the Commission.

A. The following items must be completed prior to proceeding to Operational Mode 2 (Initial Criticality).

1. Completion of significant startup punchlist items which affect the operability of the following:

   - Sampling System
   - Auxiliary Building H&V (1)
   - Emergency Feedwater System (2)
   - Plant Protective System (4)
   - Reactor Coolant System (3)
   - Waste Gas System (1)
   - Area Radiation Monitors (2)
   - Air & Gas Radiation Monitors (6)
   - Safety Injection System (2)
   - Liquid Radwaste System (4)

2. Completion of the following Preoperational Tests:

   2.083.01 Main Steam Supply and Safety Relief Valves

3. Closeout of outstanding Startup Program Test Deficiencies.

4. Approval and issuance of the following procedure:

   2.800.01 App. U Unit Load Transient Test

5. Resolution of main feedwater line break potential within the containment piping penetration room.

6. Resolution of the following items relating to radiation protection.

   a. Complete installation and calibration of health physics monitoring equipment.

   b. Complete calibration of area radiation monitors.

   c. Complete calibration of process radiation monitors.

7. Complete hanger installation.

8. Complete installation of independent DC power supplies to the series containment penetration breakers.

9. Resolution of discrepancies identified in the Facility Operating Procedures.

10. Resolution of test deficiencies relating to the failure of the Hydrogen Purge System to meet FSAR acceptance criterion. These deficiencies include:

    - Failure of the filters to pass the Freon-112 test.

    - Failure of the system to meet specified flow rate.

11. Resolution of LPSI Pump Motor Failure.

12. Resolution of loose part in safety injection system.

B. The following items must be completed prior to proceeding to Operational Mode 1 (Power Operation).

1. Completion of significant startup punchlist items which affect the operability of the following:

   - Control Room H&V (1)
   - Miscellaneous H&V (1)
   - Feedwater System (1)
   - Steam Generators (2)
   - Fuel Pool and Auxiliaries (8)
   - Waste Gas System (1)
   - Solid Radiation Waste System (4)
   - Main Steam System (2)

2. Resolution of the following outstanding operations punchlist items:

   - Instrumentation in place for CECEC Code verification

Transmitted with Am 7 12-1-78

ATTACHMENT 2 TO AMENDMENT NO. 7

LICENSE NO. NPF-6

Preoperational Tests, Startup Tests, and Other Items Which Must be Completed Prior to Proceeding To Succeeding Operational Modes

This attachment identifies certain preoperational tests, startup tests, and other items which must be completed to the Commission's satisfaction prior to proceeding to certain specified Operational Modes. Arkansas Power & Light Company shall not proceed beyond the authorized Operational Modes without prior written authorization from the Commission.

A. The following items must be completed prior to proceeding to Operational Mode 2 (Initial Criticality).

1. Completion of significant startup punchlist items which affect the operability of the following:

- Sampling System
- Auxiliary Building H&V (1)
- Emergency Feedwater System (2)
- Plant Protective System (4)
- Reactor Coolant System (3)
- Waste Gas System (1)
- Area Radiation Monitors (2)
- Air & Gas Radiation Monitors (6)
- Safety Injection System (2)
- Liquid Radwaste System (4)

2. Completion of the following Preoperational Tests:

2.083.01 Main Steam Supply and Safety Relief Valves

3. Closeout of outstanding Startup Program Test Deficiencies.

4. Approval and issuance of the following procedure:

2.800.01 App. U Unit Load Transient Test

5. Resolution of main feedwater line break potential within the containment piping penetration room.

Satisfactorily completed Proceed to Mode 2 (startup) 12-4-78

See Authorization Tab for futher info

Acceptable Completed - Proceed to Operational Mode #2 (startup)
12-4-78

6. Resolution of the following items relating to radiation protection.

   a. Complete installation and calibration of health physics monitoring equipment.

   b. Complete calibration of area radiation monitors.

   c. Complete calibration of process radiation monitors.

7. Complete hanger installation.

8. Complete installation of independent DC power supplies to the series containment penetration breakers.

9. Resolution of discrepancies identified in the Facility Operating Procedures.

10. Resolution of test deficiencies relating to the failure of the Hydrogen Purge System to meet FSAR acceptance criterion. These deficiencies include:

    - Failure of the filters to pass the Freon-112 test.

    - Failure of the system to meet specified flow rate.

11. Resolution of LPSI Pump Motor Failure.

12. Resolution of loose part in safety injection system.

13. Conformance to GDC-17 offsite power deficiencies.

14. Resolution of inverter deficiencies.

15. Chloride swipes within containment.

15. Resolution of Diesel Generator No. 2 failure.

17. Resolution of CRD-58 failure.

B. The following items must be completed prior to proceeding to Operational Mode 1 (Power Operation).

1. Completion of significant startup punchlist items which affect the operability of the following:

Acceptable Completed - Proceed to Operational Mode 1 (Power Operation)

- Control Room H&V (1)
- Miscellaneous H&V (1)
- Feedwater System (1)
- Steam Generators (2)
- Fuel Pool and Auxiliaries (8)
- Waste Gas System (1)
- Solid Radiation Waste System (4)
- Main Steam System (2)

2. Resolution of the following outstanding operations punchlist items:

   - Instrumentation in place for CECEC Code verification.