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

CONSTELLATION SOUTH TEXAS, LLC

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CITY OF AUSTIN, TEXAS

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

RENEWED FACILITY OPERATING LICENSE

Renewed License No. NPF-76

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

   A. The application for a renewed license filed by STP Nuclear Operating Company (STPNOC)*, acting on behalf of itself and for Constellation South Texas, LLC, the City Public Service Board of San Antonio (CPS), and City of Austin, Texas (COA) (the "Owners") complies with the standards and requirements of the Atomic Energy Act of 1954, as of 1954 as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the South Texas Project, Unit 1, (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-128 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

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

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

* STPNOC is authorized to act for Constellation South Texas, LLC, the City Public Service Board of San Antonio, and City of Austin, Texas and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

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

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

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

H. After weighing the environmental, economic, technical and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Renewed Facility Operating License No. NPF-76, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

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

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

2. Based on the foregoing findings by the Nuclear Regulatory Commission, Facility Operating License No. NPF-76, dated March 22, 1988, and expiring August 20, 2027, as amended, is superseded by Renewed Facility Operating License No. NPF-76 and hereby issued to STPNOC to read as follows:

A. This renewed license applies to the South Texas Project, Unit 1, a pressurized water reactor, and associated equipment (the facility) owned by Constellation South Texas, LLC, City Public Service Board of San Antonio, and City of Austin, Texas and operated by STPNOC. The facility is located in Matagorda County, Texas, west of the Colorado River, 8 miles north-northwest of the town of Matagorda and about 89 miles southwest of Houston and is described in the licensees' Final Safety Analysis Report, as supplemented and amended, and in the licensees' Environmental Report, as supplemented and amended.

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

(1) STPNOC pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in Matagorda County, Texas, in accordance with the procedures and limitations set forth in this renewed license;

(2) Constellation South Texas, LLC, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA), pursuant to the Act and 10 CFR Part 50, to possess the facility at the designated location in Matagorda County, Texas, in accordance with the procedures and limitations set forth in this renewed license;

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

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

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

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

STPNOC is authorized to operate the facility at reactor core power levels not in excess of 3,853 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 225 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SER)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Safety Parameter Display System (Section 18, SSER No. 4)*

Before startup after the first refueling outage, HL&P[**] shall perform the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to issues as described in Section 18.2 of SER Supplement 4.

(6) Supplementary Containment Purge Isolation (Section 11.5, SSER No. 4)*

HL&P shall provide, prior to startup from the first refueling outage, control room indication of the normal and supplemental containment purge sample line isolation valve position.

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

** The original licensee authorized to possess, use and operate the facility was HL&P. Consequently, historical references to certain obligations of HL&P remain in the license conditions.

(7) License Transfer

Texas Genco, LP shall provide decommissioning funding assurance, to be held in decommissioning trusts for South Texas Project, Unit 1 (Unit 1) upon the direct transfer of the Unit 1 license to Texas Genco, LP, in an amount equal to or greater than the balance in the Unit 1 decommissioning trust immediately prior to the transfer. In addition, Texas Genco, LP shall ensure that all contractual arrangements referred to in the application for approval of the transfer of the Unit 1 license to Texas Genco, LP to obtain necessary decommissioning funds for Unit 1 through a non-bypassable charge are executed and will be maintained until the decommissioning trusts are fully funded, or shall ensure that other mechanisms that provide equivalent assurance of decommissioning funding in accordance with the Commission's regulations are maintained.

(8) License Transfer

The master decommissioning trust agreement for Unit 1, at the time the direct transfer of Unit 1 to Texas Genco, LP is effected and thereafter, is subject to the following:

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

b. With respect to the decommissioning trust funds, investments in the securities or other obligations of CenterPoint Energy, Inc., or its affiliates, successors, or assigns, shall be prohibited. Except for investments in funds tied to market indices or other nonnuclear sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

c. The decommissioning trust agreement must provide that the trustee, investment advisor, or anyone else directing the investments made in the trusts shall adhere to the standards for such investments established by the Public Utility Commission of Texas (e.g., 16 Texas Administration Code § 25.301).

d. The decommissioning trust agreement must provide that except for ordinary administrative expenses, no disbursements or payments from the trusts shall be made by the trustee unless the trustee has first given the NRC 30 days prior written notice of such disbursement or payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trusts shall be made if the trustee receives prior written notice of an objection from the Director, Office of Nuclear Reactor Regulation.

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

(9) License Transfer

Texas Genco, LP shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for approval of the transfer of the Unit 1 license to Texas Genco, LP, the requirements of the Order approving the transfer, and the related safety evaluation.

(10) License Transfer

Texas Genco, LP shall provide the Director, Office of Nuclear Reactor Regulation a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from CenterPoint Energy, Inc., or its subsidiaries, to a proposed direct or indirect parent, or to any other affiliated company, facilities for the production of electric energy having a depreciated book value exceeding ten percent (10%) of such licensee's consolidated net utility plant, as recorded on Texas Genco, LP's book of accounts.

(11) Mitigation Strategy License Condition

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

a. Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

b. Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment

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

c. Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

(12) Control Room Envelope Habitability

Upon implementation of this License Amendment Request adopting TSTF 448, Revision 3, the determination of CRE unfiltered air inleakage as required by SR 4.7.7.e.3, in accordance with TS 6.8.3.q.3.(i), the assessment of CRE habitability as required by Specification 6.8.3.q.3.(ii), and the measurement of CRE pressure as required by Specification 6.8.3.q.4, shall be considered met. Following implementation:

a. For Unit 1, the first performance of SR 4.7.7.e.3, in accordance with Specification 6.8.3.q.3.(i), shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from March 9, 2004, the date of the most recent successful tracer gas test, as stated in the letter from T.J. Jordan, STP Nuclear Operating Company, to the NRC Document Control Desk, dated August 5, 2004 (NOC-AE-04001758), response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

b. For Unit 1, the first performance of the periodic assessment of CRE habitability, Specification 6.8.3.q.3.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from March 9, 2004, the date of the most recent successful tracer gas test, as stated in the letter from T.J. Jordan, STP Nuclear Operating Company, to the NRC Document Control Desk, dated August 5, 2004 (NOC-AE-04001758), response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

c. For Unit 1, the first performance of the periodic measurement of CRE pressure, Specification 6.8.3.q.4, shall be within 18 months, plus the 138 days allowed by SR 3.0.2, as measured from April 30, 2007, the date of the most recent successful pressure measurement test.

(13) License Renewal License Conditions

a. The information in the Updated Final Safety Analysis Report (UFSAR) supplement required by 10 CFR 54.21(d), as revised during the application review process, and as supplemented by Commitments (as updated by STPNOC letters submitted through May 2, 2017) applicable to South Texas Project, Unit 1, in Appendix A of the “Safety Evaluation Report Related To The License Renewal of South Texas Project, Units 1 and 2,” (SER) dated June 2017, is collectively the “License Renewal UFSAR Supplement.” This Supplement is henceforth part of the UFSAR which will be updated in accordance with 10 CFR 50.71(e). As such, STPNOC may make changes to the programs and activities applicable to South Texas Project, Unit 1, described in this Supplement provided STPNOC evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

b. This License Renewal UFSAR Supplement, as specified in License Condition (13)a above, describes programs to be implemented and activities to be completed prior to the period of extended operation.

   1. STPNOC shall implement those new programs and enhancements to existing programs no later than February 21, 2027.

   2. STPNOC shall complete those activities as noted in the Commitments applicable to South Texas Project, Unit 1, in this Supplement by February 21, 2027 or the end of the last refueling outage before the period of extended operation, whichever occurs later.

   3. STPNOC shall notify the NRC in writing within 30 days after having accomplished item b.1 above and include the status of those activities that have been or remain to be completed in item b.2 above.

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

d. Prior to August 20, 2027, destructive examinations shall be conducted on the lesser of 20 percent or 25 of the aboveground welds susceptible to loss of material due to selective leaching of aluminum bronze without backing rings and the lesser of 20 percent or 25 of the aboveground welds susceptible to loss of material due to selective leaching of aluminum bronze with backing rings. The results of the examinations shall be evaluated in accordance with the acceptance criteria, and corrective actions shall be taken when the acceptance criteria are not met, as specified in the license renewal application, as amended through supplements dated May 2, 2017, specifically as described in the license amendment supplement dated March 30, 2017.

D. Exemptions

The following exemptions are authorized by law and will not endanger life or property or the common defense and security, and certain special circumstances are present. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

(1) The facility requires a technical exemption from the requirements of 10 CFR Part 50, Appendix J, Section III.D.2(b)(ii). The justification for this exemption is contained in Section 6.2.6 of Supplement 3 to the Safety Evaluation Report. The staff's environmental assessment was published on July 2, 1987 (52 FR 25094). Therefore, pursuant to 10 CFR 50.12(a)(1), 10 CFR 50.12(a)(2)(ii) and (iii), the South Texas Project Unit 1 is hereby granted an exemption from the quoted requirement and instead, is required to perform the overall air lock leak test at pressure P_a prior to establishing containment integrity if air lock maintenance has been performed that could affect the air lock sealing capability.

(2) The facility requires a schedular exemption from the requirements of General Design Criterion 57, Appendix A to 10 CFR 50. The staff has described in detail in Supplement 4 to the Safety Evaluation Report the technical bases associated with this exemption. The staff's environmental assessment was published on June 18, 1987 (52 FR 23217). Therefore, pursuant to 10 CFR 50.12(a)(1) and 10 CFR 50.12(a)(2)(v) the South Texas Project Unit 1 is hereby granted an exemption from the requirements of GDC-57 applicable to the essential component cooling water (CCW) piping which is also used by the nonessential reactor containment building chilled water system in providing cooling to the Reactor Containment Fan Coolers (RCFC). This exemption will expire at the end of the first refueling outage.

(3) The facility was previously granted exemption from the criticality monitoring requirements of 10 CFR 70.24 (See Materials License No. SNM-1 972 dated December 29, 1986 and Section 9.1.2 of SSER No. 3). The South Texas Project Unit 1 is hereby exempted from the criticality monitoring provisions of 10 CFR 70.24 as applied to fuel assemblies held under this renewed license.

(4) The facility has been granted a schedular exemption from Section 50.71(e)(3)(i) of 10 CFR 50 to extend the date for submittal of the updated Final Safety Analysis Report to no later than one year after the date of issuance of a low power license for the South Texas Project, Unit 2. This exemption is effective until August 1990. The staff's environmental assessment was published on December 16, 1987 (52 FR 47805).

E. Fire Protection

STPNOC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report through Amendment No. 55 and the Fire Hazards Analysis Report through Amendment No. 23, and submittals dated April 29, May 7, 8 and 29, June 11, 25 and 26, 1987; February 3, March 3, and November 20, 2009; January 20, 2010; July 23, 2013; May 12 (two letters), May 19, and December 17, 2014; and as approved in the SER (NUREG-0781) dated April 1986 and its Supplements, subject to the following provision:

STPNOC may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

F. Physical Security

STPNOC shall fully implement and maintain in effect all provisions of the physical security, training and qualification, and safeguards contingency plans previously approved by the Commission and all amendments and revisions to such plans made pursuant to the authority under 10 CFR 50.90 and 10 CFR 50.54(p).

STPNOC 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 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: "South Texas Project Electric Generating Station Security, Training and Qualification, and Safeguards Contingency Plan, Revision 2" submitted by letters dated May 17 and 18, 2006.

STPNOC 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). STPNOC CSP was approved by License Amendment No. 197 and supplemented by License Amendment No. 202.

G. Not Used

H. Financial Protection

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

I. Effective Date and Expiration

This renewed license is effective as of the date of issuance and shall expire at midnight on August 20, 2047.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brian E. Holian, Acting Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A, Technical Specifications (NUREG-1305)
2. Appendix B, Environmental Protection Plan

Date of Issuance: September 28, 2017

NUREG-1346

Technical Specifications

South Texas Project, Unit Nos. 1 and 2

Docket Nos. 50-498 and 50-499

Appendix "A" to
License Nos. NPF-76 and NPF-80

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

March 1989

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

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

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

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

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a 2-section excore neutron detector.

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specification 3.6.3.

b. All equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

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

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, or reactivity control components [excluding rod cluster control assemblies (RCCAs) locked out in the integrated head package] within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT is the unit-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 Specification 6.9.1.6. Plant operation within these core operating limits is addressed within the individual Specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1.10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of injecting simulated process data where available or exercising the digital computer hardware using data base manipulation to verify OPERABILITY of alarm, interlock, and/or trip functions.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same Committed Effective Dose Equivalent dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The Committed Effective Dose Equivalent dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1988 (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).

DOSE EQUIVALENT XE-133

1.12 DOSE EQUIVALENT XE-133 shall be the 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."

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) 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.

FREQUENCY NOTATION

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

GASEOUS WASTE PROCESING SYSTEM

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

IDENTIFIED LEAKAGE

1.16 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 Coolant System.

MASTER RELAY TEST

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

MEMBER(S) OF THE PUBLIC

1.18 MEMBER(S) OF THE PUBLIC means an individual in a controlled area or UNRESTRICTED AREA. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

1.19 Not Used

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

1.23 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube 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.

1.24 Not Used

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3,853 Mwt.

REACTOR TRIP SYSTEM RESPONSE TIME

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

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

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

SLAVE RELAY TEST

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.35 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

VENTING

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

TABLE 1.1

FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL MODES

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

*Excluding decay heat.

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in the Core Operating Limits Report.

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for the WRB-1 DNB correlation and $\geq$ 1.14 for the WRB-2M DNB correlation.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58 °F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of 10 CFR 50.36(c)(1).

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of 10 CFR 50.36(c)(1).

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of 10 CFR 50.36(c)(1).

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2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Value column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1.	Manual Reactor Trip	N.A.	N.A.
2.	Power Range, Neutron Flux		
	a. High Setpoint	≤109% of RTP**	≤110.7% of RTP**
	b. Low Setpoint	≤25% of RTP**	≤27.7% of RTP**
3.	Power Range, Neutron Flux, High Positive Rate	≤5% of RTP** with a time constant ≥2 seconds	≤6.7% of RTP** with a time constant ≥2 seconds
4.	Deleted		
5.	Intermediate Range, Neutron Flux	≤25% of RTP**	≤31.1% of RTP**
6.	Source Range, Neutron Flux	≤10^5 CPS	≤1.4×10^5 cps
7.	Overtemperature ΔT	See Note 1	See Note 2
8.	Overpower ΔT	See Note 3	See Note 4
9.	Pressurizer Pressure-Low	≥1870 psig	≥1860 psig
10.	Pressurizer Pressure-High	≤2380 psig	≤2390 psig
11.	Pressurizer Water Level-High	≤92% of instrument span	≤94.1% of instrument span
12.	Reactor Coolant Flow-Low	≥91.8% of loop design flow*	≥91.4% of loop design flow*

* Loop design flow = As specified in the Core Operating Limits Report
** RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low	≥ 20.0% of narrow range instrument span	≥ 18.0% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	≥ 10,014 volts	≥ 9339 volts
15. Underfrequency - Reactor Coolant Pumps	≥ 57.2 Hz	≥ 57.1 Hz
16. Turbine Trip		
a. Low Emergency Trip Fluid Pressure	≥ 1245.8 psig	≥ 1114.5 psig
b. Turbine Stop Valve Closure	< Fully closed	Fully closed
17. Safety Injection Input from ESFAS	N.A.	N.A.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	≤10% of RTP**	≤11.7% of RTP**
2) P-13 input	≤10% RTP** Turbine Inlet Pressure Equivalent	≤11.7% RTP** Turbine Inlet Pressure Equivalent
c. Power Range Neutron Flux, P-8	≤40% of RTP**	≤41.7% of RTP**
d. Power Range Neutron Flux, P-9	≤50% of RTP**	≤51.7% of RTP**
e. Power Range Neutron Flux, P-10	≥10% of RTP**	≥8.3% of RTP**
f. Turbine Inlet Pressure, P-13	≤10% RTP** Turbine Inlet Pressure Equivalent	≤11.7% RTP** Turbine Inlet Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.

**RTP = RATED THERMAL POWER

T[illegible].2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT = Measured ΔT by RCS Instrumentation;

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

τ_1, τ_2 = Time constant utilized in lead-lag compensator for ΔT, *

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

τ_3 = Time constant utilized in the lag compensator for ΔT, *

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

K_1 = *

K_2 = *

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

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, *

T = Measured RCS Average temperature, °F;

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

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, *

* : As specified in the Core Operating Limits Report.

TAB[illegible] 1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T' = Nominal T_{avg} at RATED THERMAL POWER as specified in the Core Operating Limits Report;

K_3 = *

P = Measured Pressurizer pressure, psig;

P' = Nominal RCS operating pressure as specified in the Core Operating Limits Report;

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

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests. The $f_1(\Delta I)$ value is as specified in the Core Operating Limits Report.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.6% ΔT span.

*: As specified in the Core Operating Limits Report.

T[illegible].2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where:

ΔT = As defined in Note 1,

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

τ_1, τ_2 = *

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

τ_3 = *

ΔT_0 = As defined in Note 1,

K_4 = *

K_5 = *

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

τ_7 = Time constant utilized in the rate-lag compensator for T_{avg} , *

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

τ_6 = *

* : As specified in the Core Operating Limits Report.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = *

T = Measured RCS Average Temperature, °F,

T" = Nominal T_{avg} at RATED THERMAL POWER as specified in the Core Operating Limits Report,

S = As defined in Note 1, and

$f_2(\Delta I)$ = *

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% ΔT span.

* : As specified in the Core Operating Limits Report.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4. 0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

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

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

This specification is not applicable in MODE 5 or 6.

3.0.4 When a Limiting Condition of Operation is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

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

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

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

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

LIMITING CONDITION FOR OPERATION (Continued)

3.0.5 Limiting Conditions for Operation including the associated ACTION requirements shall apply to each unit individually unless otherwise indicated as follows:

a. Whenever the Limiting Conditions for Operation refers to systems or components which are shared by both units, the ACTION requirements will apply to both units simultaneously.

b. Whenever the Limiting Conditions for Operation applies to only one unit, this will be identified in the APPLICABILITY section of the specification; and

c. Whenever certain portions of a specification contain operating parameters, Setpoints, etc., which are different for each unit, this will be identified in parentheses, footnotes or body of the requirement.

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

APPLICABILITY

SURVEILLANCE REQUIREMENTS

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

Failure to meet a Surveillance Requirement, whether such a failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Conditions for Operation. Failure to perform a Surveillance within the specified surveillance interval shall be failure to meet the Limiting Conditions for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

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

4.0.3 If it is discovered that a Surveillance was not performed within its specified surveillance interval (including the allowed extension per Specification 4.0.2), then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. This delay period is permitted to allow performance of the surveillance. 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 Limiting Condition for Operation must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met and the applicable Condition(s) must be entered.

4.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the Surveillance Requirement(s) associated with the LCO have been met within their specified Frequency, except as provided by Specification 4.0.3. When an LCO is not met due to Surveillance Requirement(s) not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

a. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves, and inservice inspection of ASME Code Class 1,2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(f) and Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(f)(6)(i) or Section 50.55a(g)(6)(i), or where the component has been found to qualify for exemption from special treatment;

SURVEILLANCE REQUIREMENTS (Continued)

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

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

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

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

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

4.0.6 Surveillance Requirements shall apply to each unit individually unless otherwise indicated as stated in Specification 3.0.5 for individual specifications or whenever certain portions of a specification contain surveillance parameters different for each unit, which will be identified in parentheses, footnotes or body of the requirement.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits provided in the CORE OPERATING LIMITS REPORT (COLR).

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

ACTION:

With the SHUTDOWN MARGIN not within the limit initiate boration within 15 minutes and continue boration until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

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

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

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

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

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. When in MODE 3, 4, or 5, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1d., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. The provisions of Specification 4.0.4 are not applicable.

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3.1.1.2 (This specification not used)

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REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the Core Operating Limits Report (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-2a.

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

ACTION:

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

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

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

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

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

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

**See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

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

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

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

* Measurement of the MTC in accordance with Surveillance Requirement 4.1.1.3.b may be suspended, provided that the benchmark criteria in WCAP-13749-P-A (refer to 6.9.1.6.b.10) and the Revised Prediction specified in the COLR are satisfied.

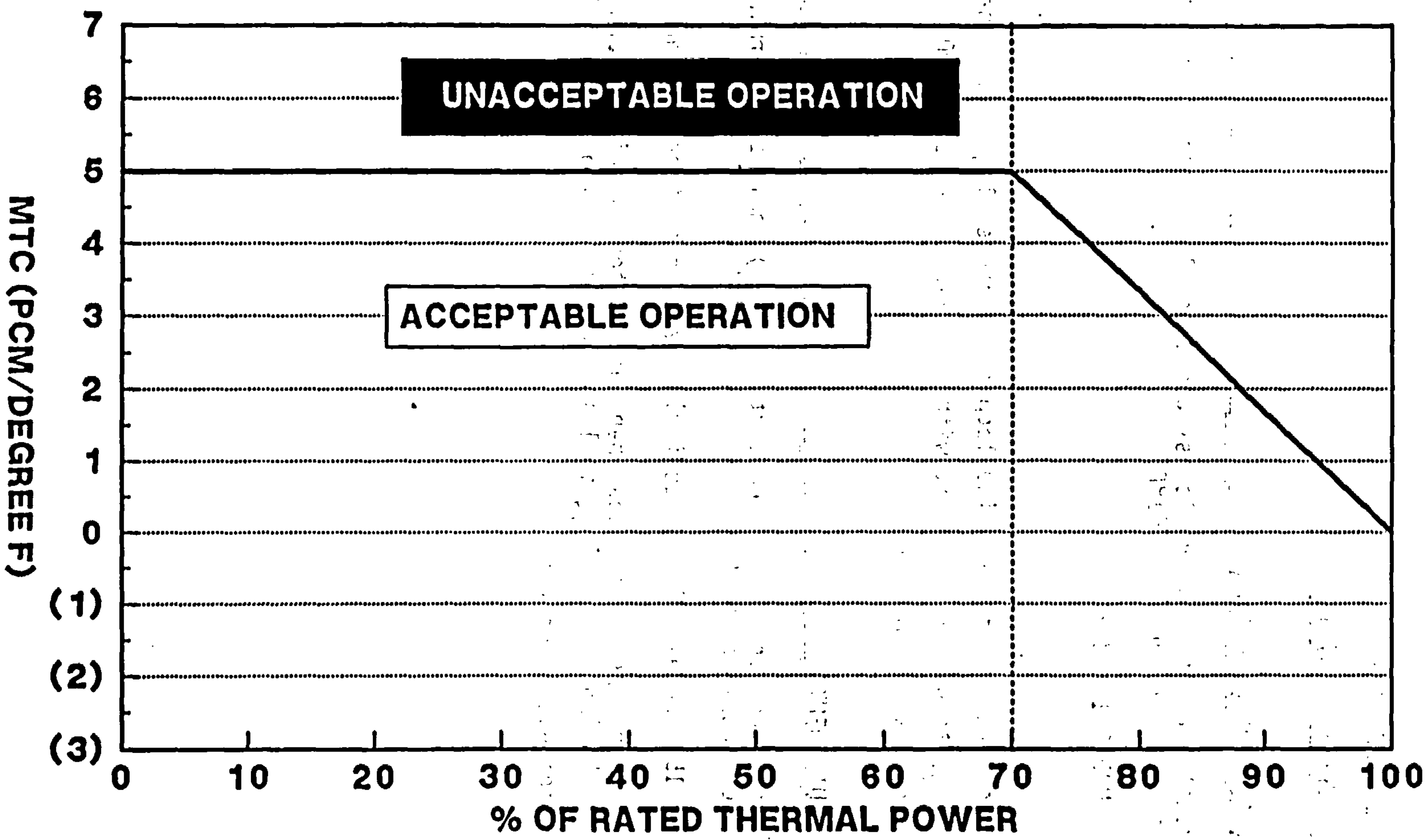

Figure 3.1-2a

MTC versus Power Level

NOTE: Cycle specific MTC limits are displayed in the COLR.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 561°F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 561°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 561°F:

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

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 571°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

---

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

**See Special Test Exceptions Specification 3.10.3.

3/4.1.2 (This specification not used)

Pages 3/4 1-10 through 3/4 1-15 have been deleted.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

b. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

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

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits as specified in the Core Operating Limits Report (COLR). The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

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

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

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

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

ACTION (Continued)

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

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

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits as specified in the COLR. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

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

Single Rod Cluster Control Assembly Withdrawal at Full Power

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

Major Secondary Coolant System Pipe Rupture

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

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

3.1.3.3 (This specification not used)

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.8 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 561°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. At a frequency in accordance with the Surveillance Frequency Control Program.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn, as specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

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

b. At a frequency in accordance with the Surveillance Frequency Control Program.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
**With K_{eff} greater than or equal to 1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AFD outside of the above required target band, and with THERMAL POWER:

1. greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   a) Restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

2. greater than or equal to 50%, but less than 90% of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the target band for more than 1 hour cumulative penalty deviation during the previous 24 hours, and

      2) The indicated AFD is within the Acceptable Operation limits specified in the COLR.

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

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1., provided that the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the Target Band during this testing without penalty deviation.

*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

3. greater than 15%, but less than 50% of RATED THERMAL POWER:

   THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour cumulative penalty deviation during the previous 24 hours.

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the target band, and the indicated AFD has not been outside of the target band for more than 1 hour cumulative penalty deviation during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

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

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

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

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

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference

SURVEILLANCE REQUIREMENTS (Continued)

pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the predicted value at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

FIGURE 3.2-1 HAS BEEN DELETED

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq (F_Q^{RTP}/P)^* K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq (F_Q^{RTP}/0.5)^* K(Z) \text{ for } P \leq 0.5$$

Where: F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the Core Operating Limits Report (COLR).

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

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

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

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

FIGURE 3.2-2 HAS BEEN DELETED

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Obtaining a core power distribution measurement,

b. Increasing the measured F_{xy} by the applicable manufacturing and measurement uncertainties as specified in the COLR,

c. Comparing the F_{xy} computed (F_{xy}^{C}) obtained in Specification 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP}[1+PF_{xy}(1-P)],$$

Where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier F_{xy} specified in the COLR, and P is the fraction of RATED THERMAL POWER at which F_{xy} is measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional core power distribution measurements shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} either:

a) Within 24 hours after exceeding by 20% RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

b) At least once per 31 Effective Full Power days (EFPD), whichever occurs first.

2) When F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional core power distribution measurements shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits used in the Constant Axial Offset Control analysis for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes as specified in the COLR per Specification 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e, above, are not applicable in the core plane regions specified in the COLR.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_{Q}(Z)$ shall be evaluated to determine if $F_{Q}(Z)$ is within its limits.

4.2.2.3 When $F_{Q}(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_{Q}(Z)$ shall be obtained from a core power distribution measurement and increased by the applicable manufacturing and measurement uncertainties as specified in the COLR.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be less than $F^{RTP}_{\Delta H}$ [1.0 +$PF_{\Delta H}$ (1-P)]

Where: $F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ Limit at RATED THERMAL POWER (RTP) specified in the Core Operating Limits Report (COLR).

$PF_{\Delta H}$ = the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the Core Operating Limits Report (COLR).

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

APPLICABILITY: MODE 1.

ACTION:

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

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

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the limit required by ACTION a., above; THERMAL POWER may then be increased, provided $F^N_{\Delta H}$ is demonstrated through core power distribution measurement to be within its limits.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be demonstrated to be within its limit prior to operation above 75% RATED THERMAL POWER after each fueling loading and at least once per 31 EFPD thereafter by:

a. Obtaining a core power distribution measurement.

b. Increasing the measured value of $F^N_{\Delta H}$ by the applicable measurement uncertainty as specified in the COLR.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION: With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by measuring core power distribution to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by using:

a. The Power Distribution Monitoring System (PDMS), or

b. The movable incore detectors by either:

1. Using the four pairs of symmetric thimble locations, or

2. Using the movable incore detection system to monitor the QUADRANT POWER TILT RATIO with a full incore map.

* See Special Test Exceptions Specification 3.10.2.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System $T_{avg} \leq$ the limit as specified in the Core Operating Limits Report

b. Pressurizer Pressure, > the limit as specified in the Core Operating Limits Report

c. Thermal Design Reactor Coolant System Flow, $\geq$ 392,000 gpm

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at a frequency in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable for verification that RCS flow is within its limit.

4.2.5.2 The RCS flow rate indicators shall be subjected to a channel calibration at a frequency in accordance with the Surveillance Frequency Control Program.

> NOTE
> SR 4.2.5.3 is required at beginning-of-cycle with reactor power $\geq$ 90% RTP.

4.2.5.3 The RCS total flow rate shall be determined by precision heat balance or elbow tap ΔP measurements at a frequency in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Chapter 16 in the Updated Final Safety Analysis Report (UFSAR).

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1 and at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-1.

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

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
7. Extended Range, Neutron Flux	2	0	2	3, 4, 5	5
8. Overtemperature ΔT	4	2	3	1, 2	6
9. Overpower ΔT	4	2	3	1, 2	6
10. Pressurizer Pressure -- Low (Interlocked with P-7)	4	2	3	1	6
11. Pressurizer Pressure-- High	4	2	3	1, 2	6
12. Pressurizer Water Level--High (Interlocked with P-7)	4	2	3	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6
14. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6
15. Undervoltage--Reactor Coolant Pumps (Interlocked with P-7)	4-1/bus	2	3	1	6
16. Underfrequency--Reactor Coolant Pumps (Interlocked with P-7)	4-1/bus	2	3	1	6
17. Turbine Trip (Interlocked with P-9)					
a. Low Emergency Trip Fluid Pressure	3	2	2	1	6
b. Turbine Stop Valve Closure	4	2	3	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Safety Injection Input from ESFAS	2	1	2	1, 2	9A
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	8
f. Turbine Inlet Pressure, P-13	2	1	2	1	8
20. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10, 12A

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
21. Automatic Trip and Interlock Logic	2	1	2	1, 2	9A
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

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

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

ACTION STATEMENTS

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

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

a. For Functional Units with installed bypass test capability,

   Note: A channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.1.1, provided no more than one channel is in bypass at any time.

   1. The inoperable channel may be placed in bypass, and must be placed in the tripped condition within 72 hours, and

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

b. For Functional Units with no installed bypass test capability,

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

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

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

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

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

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

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes. Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 72 hours, or immediately suspend all operations involving positive reactivity changes.

Note: Plant temperature changes or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

b. With the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement,

Immediately suspend all operations involving positive reactivity changes,

AND

Within 15 minutes isolate unborated water flow paths from the reactor makeup water system to the reactor coolant system,

AND

Perform either of the following:

Restore at least one channel to OPERABLE status within 1 hour,

OR

1. Within 2 hours secure each unborated water flow path to the reactor coolant system by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured,

   AND

2. Within 4 hours and once per 12 hours thereafter, verify SHUTDOWN MARGIN is within limits.

Note: Operations involving plant temperature changes may proceed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

a. For Functional Units with installed bypass test capability, the inoperable channel may be placed in bypass, and must be placed in the tripped condition within 72 hours.

   Note: A channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.1.1, provided no more than one channel is in bypass at any time.

b. For Functional Units with no installed bypass test capability,

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

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

ACTION 7 - (Not Used)

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

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

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

b. With the number of OPERABLE channels more than one less then the Minimum Channels OPERABLE requirement, within 1 hour restore at least one inoperable channel to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status or initiate action to fully insert all rods within 48 hours, and place the rod control system in a condition incapable of rod withdrawal within the next hour.

ACTION 11 - (Not Used)

ACTION 12 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours, or be in at least HOT STANDBY within the next 6 hours. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 12A - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, within 48 hours restore it to OPERABLE status or initiate action to fully insert all rods, and within the next hour place the rod control system in a condition incapable of rod withdrawal.

TABLE 3.3-2

(This table number not used)

Correction letter of 5-25-93

TABLE 3.3-2 (Continued)

(This table number not used)

Correction letter of 5-25-93

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(20)	CHANNEL CALIBRATION(20)	ANALOG CHANNEL OPERATIONAL TEST (19)(20)	TRIP ACTUATING DEVICE OPERATIONAL TEST(20)	ACTUATION LOGIC TEST(20)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A	(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint		(2, 4), (3, 4), (4, 6), (4, 5)	(17)	N.A.	N.A.	1, 2
b. Low Setpoint		(4)	S/U (1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	(4)	(17)	N.A.	N.A.	1, 2
4. Deleted						
5. Intermediate Range, Neutron Flux		(4, 5)	S/U (1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux		(4, 5)	S/U (1), (9) (17)	N.A.	N.A.	2**, 3, 4, 5
7. Extended Range, Neutron Flux		(4)	(12, 17)	N.A.	N.A.	3, 4, 5
8. Overtemperature ΔT			(17)	N.A.	N.A.	1, 2
9. Overpower ΔT			(17)	N.A.	N.A.	1, 2
10. Pressurizer Pressure--Low			(17)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(20)	CHANNEL CALIBRATION(20)	ANALOG CHANNEL OPERATIONAL TEST (19)(20)	TRIP ACTUATING DEVICE OPERATIONAL TEST(20)	ACTUATION LOGIC TEST(20)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. Pressurizer Pressure --High			(17)	N.A.	N.A.	1, 2
12. Pressurizer Water Level --High			(17)	N.A.	N.A.	1
13. Reactor Coolant Flow --Low			(17, 18)	N.A.	N.A.	1
14. Steam Generator Water Level--Low-Low			(17, 18)	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.		N.A.	(17)	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.		N.A.	(17)	N.A.	1
17. Turbine Trip						
a. Low Emergency Trip Fluid Pressure	N.A.		N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.		N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESFAS	N.A.	N.A.	N.A.		N.A.	1, 2

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (20)	CHANNEL CALIBRATION(20)	ANALOG CHANNEL OPERATIONAL TEST (19)(20)	TRIP ACTUATING DEVICE OPERATIONAL TEST(20)	ACTUATION LOGIC TEST(20)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	(4)		N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	(4)		N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	(4)		N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	(4)		N.A.	N.A.	1
e. Power Range Neutron Flux, P-10	N.A.	(4)		N.A.	N.A.	1, 2
f. Turbine Inlet Pressure, P-13	N.A.			N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	(7, 11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	(7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	(15), R(16)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

(1) If not performed in previous 31 days.

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

*(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, monthly shall mean at least once per 31 EFPD.

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

(5) Detector plateau curves shall be obtained and evaluated. If a low noise preamplifier is used with the Source Range Detector, no plateau curve is obtained. Instead, with the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

*(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, quarterly shall mean at least once per 92 EFPD.

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

(8) (Not Used)

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

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

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

(12) OPERABILITY shall be verified by a check of memory devices, input accuracies, Boron Dilution Alarm setpoints, output values, and software functions.

(13) (Not used)

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

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

(16) Automatic undervoltage trip.

(17) Each channel shall be tested at a frequency in accordance with the Surveillance Frequency Control Program.

(18) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

(19) For channels with bypass test instrumentation, input relays are tested at a frequency in accordance with the Surveillance Frequency Control Program.

(20) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-1.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Chapter 16 in the UFSAR.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

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

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2 and at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-2.

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

a. Each logic train is verified at a frequency in accordance with the Surveillance Frequency Control Program,

b. Each actuation train is verified at a frequency in accordance with the Surveillance Frequency Control Program*, and

c. One channel per function so that all channels are verified at least once per N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

*If an ESFAS instrumentation channel is inoperable due to response times exceeding the required limits, perform an engineering evaluation to determine if the verification failure is a result of degradation of the actuation relays. If degradation of the actuation relays is determined to be the cause, increase the ENGINEERED SAFETY FEATURES RESPONSE TIME surveillance frequency such that all trains are verified at a frequency specified in the Surveillance Frequency Control Program.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
c. Actuation Relays	3	2	3	1, 2, 3, 4	14
d. Containment Pressure--High-1	3	2	2	1, 2, 3, 4	20A
e. Pressurizer Pressure--Low	4	2	3	1, 2, 3#	20
f. Compensated Steam Line Pressure–Low	3/steam line	2/steam line any steam line	2/steam line in each steam line	1, 2, 3#	20

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	19
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
c. Actuation Relays	3	2	3	1, 2, 3, 4	14
d. Containment Pressure--High-3	4	2	3	1, 2, 3	17
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	19
2) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
3) Actuation Relays	3	2	3	1, 2, 3, 4	14
4) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
b. Containment Ventilation Isolation					
1) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	18
2) Actuation Relays***	3	2	3	1, 2, 3, 4	18
3) Safety Injection ***	See Item 1. above for all Safety Injection initiating functions and requirements.				
4) RCB Purge Radioactivity- High	2	1	2	1, 2, 3, 4	18
5) Containment Spray- Manual Initiation	See Item 2. above for Containment Spray manual initiating functions and requirements.				
6) Phase "A" Isolation- Manual Isolation	See Item 3.a. above for Phase "A" Isolation manual initiating functions and requirements.				
c. Phase "B" Isolation					
1) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
2) Actuation Relays	3	2	3	1, 2, 3, 4	14
3) Containment Pressure -- High-3	4	2	3	1, 2, 3	17
4) Containment Spray-- Manual Initiation	See Item 2. above for Containment Spray manual initiating functions and requirements.				
d. RCP Seal Injection Isolation					
1) Automatic Actuation Logic and Actuation Relays	1	1	1	1, 2, 3, 4	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3.d. RCP Seal Injection Isolation (Continued)					
2) Charging Header Pressure - Low	1	1	1	1, 2, 3, 4	16
Coincident with Phase "A" Isolation	See item 3.a. above for Phase "A" Isolation initiating functions and requirements				
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	2/steam line	1/steam line	2/operating steam line	1, 2, 3	24
2) System	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Steam Line Pressure - Negative Rate--High	3/steam line	2/steam line any steam line	2/ steam line in each steam line	3###	20
d. Containment Pressure - High-2	3	2	2	1, 2, 3	20
e. Compensated Steam Line Pressure - Low	3/steam line	2/steam line any steam line	2/steam line in each steam line	1, 2, 3#	20

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	25
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	20
c. Deleted					
d. Deleted					
e. Safety Injection	See Item 1. for all Safety Injection initiating functions and requirements.				
f. T_{avg}-Low coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	4 (1/loop)	2	3	1, 2, 3	20

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	1/pump	1/pump	1/pump	1, 2, 3	26
b. Automatic Actuation Logic	2	1	2	1, 2, 3	22
c. Actuation Relays	3	2	3	1, 2, 3	22
d. Stm. Gen. Water Level -- Low-Low Start Motor-Driven Pumps and Turbine-Driven Pump	4 stm. gen.	2 stm. gen. in any stm. gen.	3/stm. gen. in each stm. gen.	1, 2, 3	20
e. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
f. Loss of Power (Motor Driven Pumps Only)	See Item 8. below for all Loss of Power initiating functions and requirements.				
7. Automatic Switchover to Containment Sump****					
a. Automatic Actuation Logic and Actuation Relays	3-1/train	1/train	1/train	1, 2, 3, 4	19A
b. RWST Level -- Low-Low	3-1/train	1/train	1/train	1, 2, 3, 4	19A
Coincident With: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Loss of Power					
a. 4.16 kV ESF Bus Under-voltage-Loss of Voltage	4/bus	2/bus	3 /bus	1, 2, 3, 4	20A
b. 4.16 kV ESF Bus Under-voltage-Tolerable Degraded Voltage Coincident with SI	4/bus	2/bus	3/bus	1, 2, 3, 4	20A
c. 4. 16 kV ESF Bus Under-voltage - Sustained Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20A
9. Engineered Safety Features					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	1	2	1, 2, 3	23

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Control Room Ventilation					
a. Manual Initiation	3 (1/train)	2 (1/train)	3 (1/train)	1, 2, 3, 4	27
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Automatic Actuation Logic and Actuation Relays	3	2	3	1, 2, 3, 4	27
d. Control Room Intake Air Radioactivity - High	2	1	2	1, 2, 3, 4	28
e. Loss of Power	See Item 8. above for all Loss of Power initiating functions and requirements.				

TABLE 3.3-3 (Continued)
TABLE NOTATIONS

***Function is actuated by either actuation train A or actuation train B. Actuation train C is not used for this function.

****Automatic switchover to containment sump is accomplished for each train using the corresponding RWST level transmitter.

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

Trip function automatically blocked above P-11 and may be blocked below P-11 when Low Compensated Steamline Pressure Protection is not blocked.

ACTION STATEMENTS

ACTION 14 - NOTE: The provision to apply the CRMP does not apply for Item 3.a.2, Containment Isolation Phase A Automatic Actuation Logic; Item 3.a.3. Containment Isolation Phase A Actuation Relays; Item 3.c.1 Containment Isolation Phase B Automatic Actuation Logic; or Item 3.c.2, Containment Isolation Phase B Actuation Relays.

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

b. With the number of OPERABLE channels more than one less than the Minimum Channels OPERABLE requirement, within 1 hour restore at least one inoperable channel to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 15 - (Not Used)

ACTION 16 - With the Charging Header Pressure channel inoperable:

a) Place the Charging Header Pressure channel in the tripped condition within one hour and

b) Restore the Charging Header Pressure channel to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 17 - NOTE: The provision to apply the CRMP does not apply for Item 3.c.3, Containment Isolation Phase B on Containment Pressure Hi-3.

a. With the number of OPERABLE channels one less than the Total Number of Channels, within 72 hours place the inoperable channel in the bypassed condition or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1.

b. With the number of OPERABLE channels more than one less than the Total Number of Channels, within 1 hour apply the requirements of the CRMP or be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours. This action is not required for the surveillance testing provision of Action 17a.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 18 - a) With less than the Minimum Channels OPERABLE requirement for Automatic Actuation Logic or Actuation Relays, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

b) MODE 1, 2, 3, or 4

1. With one less than the Minimum Channels OPERABLE requirement for RCB Purge Radioactivity-High, within 30 days restore the inoperable channel or maintain the containment purge supply and exhaust valves closed.

NOTE:
MODE 1, 2, 3, or 4: Supplementary containment purge supply and isolation valves may be open during the allowed outage time for up to 2 hours at a time for required purge operation provided the valves are under administrative control.

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2. With two less than the Minimum Channels OPERABLE requirement for RCB Purge Radioactivity-High, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

ACTION 19 - NOTE: The provision to apply the CRMP does not apply for Item 3.a.1, Containment Isolation Phase A Manual Isolation.

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

b. With the number of OPERABLE channels more than one less than the Minimum Channels OPERABLE requirement, within 1 hour restore at least one channel to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19A a. With one train with the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, within 48 hours restore the inoperable channel to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With more than one train with the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, within 1 hour restore the channels for at least two trains to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-3 (Continued)
ACTION STATEMENTS (Continued)

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. For Functional Units with installed bypass test capability, the inoperable channel may be placed in bypass, and must be placed in the tripped condition within 72 hours.

Note: A channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1, provided no more than one channel is in bypass at any time.

b. For Functional Units with no installed bypass test capability,

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

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

ACTION 20A - a. With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

For Functional Units with installed bypass test capability, the inoperable channel may be placed in bypass, and within 72 hours place the channel in the tripped condition or apply the requirements of the CRMP.

Note: A channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1, provided no more than one channel is in bypass at any time.

For Functional Units with no installed bypass test capability,

1. Within 72 hours place the inoperable channel in the tripped condition or apply the requirements of the CRMP, and

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

b. With the number of OPERABLE channels more than one less than the Total Number of Channels, within 1 hour restore at least two channels to OPERABLE status for functions with three channels and restore at least 3 channels to OPERABLE status for functions that have four channels, or apply the requirements of the CRMP; or be in at least HOT STANDBY within the next 6 hours and be in at least HOT SHUTDOWN within the following 6 hours, and be in COLD SHUTDOWN within the following 24 hours. This action is not required for the surveillance testing provision in the note to Action 20A.a.

ACTION 21 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

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

b. With the number of OPERABLE channels more than one less than the Minimum Channels OPERABLE requirement, within one hour restore at least one channel to OPERABLE status for functions with two channels or restore at least two channels to OPERABLE status for functions with three channels, or apply the requirements of the CRMP; or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

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

ACTION 26- With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement, declare the affected Auxiliary Feed Water Pump inoperable and take ACTION required by Specification 3.7.1.2.

ACTION 27- For an inoperable channel, declare its associated ventilation train inoperable and apply the actions of Specification 3.7.7.

ACTION 28 -

a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 7 days initiate and maintain operation of the Control Room Makeup and Cleanup Filtration System (at 100% capacity) in the recirculation and makeup filtration mode.

b. With the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement, within 12 hours initiate and maintain operation of the Control Room Makeup and Cleanup Filtration System (at 100% capacity) in the recirculation and makeup filtration mode.

c. With required ACTION 28a. or 28b. not met in MODE 1, 2, 3, or 4, be in MODE 3 in 6 hours and in MODE 5 in the following 30 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.
d. Containment Pressure--High 1	≤ 3.0 psig	≤ 4.0 psig
e. Pressurizer Pressure--Low	≥ 1857 psig	≥ 1851 psig
f. Compensated Steam Line Pressure-Low	≥ 735 psig	≥ 709 psig*
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.
d. Containment Pressure--High-3	≤ 9.5 psig	≤ 10.5 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT			TRIP SETPOINT	ALLOWABLE VALUE
3.	Containment Isolation			
	a.	Phase "A" Isolation		
		1) Manual Initiation	N.A.	N.A.
		2) Automatic Actuation Logic	N.A.	N.A.
		3) Actuation Relays	N.A.	N.A.
		4) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
	b.	Containment Ventilation Isolation		
		1) Automatic Actuation Logic	N.A.	N.A.
		2) Actuation Relays	N.A.	N.A.
		3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
		4) RCB Purge Radioactivity-High	$\leq 5\times10^{-4}$ ### μCi/cc	$\leq 6.4\times10^{-4}$ μCi/cc
		5) Containment Spray - Manual Initiation	See Item 2. above for Containment Spray manual initiation Trip Setpoints and Allowable Values.	
		6) Phase "A" Isolation - Manual Initiation	See Item 3.a. above for Phase "A" Isolation manual initiation Trip Setpoints and Allowable Values.	
	c.	Phase "B" Isolation		
		1) Automatic Actuation Logic	N.A.	N.A.
		2) Actuation Relays	N.A.	N.A.
		3) Containment Pressure--High-3	$\leq$ 9.5 psig	$\leq$ 10.5 psig
		4) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiation Trip Setpoints and Allowable Values.	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT			TRIP SETPOUNT	ALLOWABLE VALUE
	d.	RCP Seal Injection Isolation		
	1)	Automatic Actuation Logic and Activation Relays	N.A.	N.A.
	2)	Charging Header Pressure – Low	≥ 560.0 psig	≥ 495.4 psig
		Coincident with Phase "A" Isolation	See Item 3.a. above for Phase "A" Isolation Setpoints and Allowable Values	
4.		Steam Line Isolation		
	a.	Manual Initiation	N.A.	N.A.
	b.	Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
	c.	Steam Line Pressure - Negative Rate--High	≤ 100 psi	≤ 126 psi**
	d.	Containment Pressure - High-2	≤ 3.0 psig	≤ 4.0 psig
	e.	Compensated Steam Line Pressure - Low	≥ 735 psig	≥ 709 psig*
5.		Turbine Trip and Feedwater Isolation		
	a.	Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
	b.	Steam Generator Water Level--High-High (P-14)	≤ 87.5% of narrow range instrument span.	≤ 89.8% of narrow range instrument span.
	c.	Deleted		

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)		
d. Deleted		
e. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
f. T_{avg} - Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	≥ 574°F	≥ 571.7°F
6. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.
d. Steam Generator Water Level--Low-Low	≥ 20.0% of narrow range instrument span	≥ 18.0% of narrow range instrument span
e. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)		
f. Loss of Power (Motor Driven Pumps Only)	See Item 8. below for all Loss of Power Trip Setpoints and Allowable Values.	
7. Automatic Switchover to Containment Sump		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. RWST Level--Low-Low	$\geq$ 11%	$\geq$ 9.1%
Coincident With: Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
8. Loss of Power		
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	$\geq$ 3107 volts with a $\leq$ 1.75 second time delay.	$\geq$ 2979 volts with a $\leq$ 1.93 second time delay.
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	$\geq$ 3835 volts with a $\leq$ 35 second time delay.	$\geq$ 3786 volts with a $\leq$ 39 second time delay.
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	$\geq$ 3835 volts with a $\leq$ 50 second time delay.	$\geq$ 3786 volts with a $\leq$ 55 second time delay.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
9.	Engineered Safety Features Actuation System Interlocks		
	a. Pressurizer Pressure, P-11	$\leq$ 1985 psig	$\leq$ 1995 psig
	b. Low-Low T_{AVG}, P-12	$\geq$ 563°F	$\geq$ 560.7°F
	c. Reactor Trip, P-4	N.A.	N.A.
10.	Control Room Ventilation		
	a. Manual Initiation	N.A.	N.A.
	b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
	c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
	d. Control Room Intake Air Radioactivity – High	$\leq 6.1\times10^{-5}$ μCi/cc	$\leq 7.8\times10^{-5}$ μCi/cc
	e. Loss of Power	See Item 8. above for all Loss of Power Trip Setpoints and Allowable Values	

(This page not used)

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

* Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

** The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

\# Deleted

\## Deleted

\### This setpoint value may be increased up to the equivalent limits of ODCM Control 3.11.2.1 in accordance with the methodology and parameters of the ODCM during containment purge or vent for pressure control, ALARA and respirable air quality considerations for personnel entry.

TABLE 3.3-5

(This table number not used)

Correction letter of 5-25-93

TABLE 3.3-5 (Continued)

(This table number not used)

Correction letter of 5-25-93

TABLE 3.3-5 (Continued)

(This table number not used)

Correction letter of 5-25-93

TABLE 3.3-5 (Continued)

(This table number not used)

Correction letter of 5-25-93

TABLE 3.3-5 (Continued)

(This table number not used)

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)								
a. Manual Initiation	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4
d. Containment Pressure-High-1				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
e. Pressurizer Pressure-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Compensated Steam Line Pressure-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4
d. Containment Pressure-High-3				N.A.	N.A.	N.A.	N.A.	1, 2, 3
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
3) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4
4) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Containment Ventilation Isolation								
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) RCB Purge Radioactivity-High				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
5) Containment Spray - Manual Initiation	See Item 2. above for Containment Spray manual initiation Surveillance Requirements.							
6) Phase "A" Isolation-Manual Initiation	See Item 3. a. above for Phase "A" Isolation manual initiation Surveillance Requirements.							
c. Phase "B" Isolation								
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3, 4
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3, 4
3) Containment Pressure--High-3				N.A.	N.A.	N.A.	N.A.	1, 2, 3
4) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiation Surveillance Requirements.							
d. RCP Seal Injection Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.		(8)	1, 2, 3, 4
2) Charging Header Pressure - Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Coincident with Phase "A" Isolation	See Item 3.a. above for Phase "A" surveillance requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	(1)	(6)	(8)	1, 2, 3
c. Steam Line Pressure-Negative Rate-High				N.A.	N.A.	N.A.	N.A.	3
d. Containment Pressure - High-2				N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Compensated Steam Line Pressure-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	(1)	(6)	(8)	1, 2, 3
b. Steam Generator Water Level-High-High (P-14)				N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Deleted								
d. Deleted								
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Turbine Trip and Feedwater Isolation (Continued)								
f. Tavg -Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)				N.A.	N.A.	N.A.	N.A.	1, 2, 3
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	(1)	N.A.	N.A.	1, 2, 3
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(6)	(8)	1, 2, 3
d. Steam Generator Water Level--Low-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Loss of Power	See Item 8. below for all Loss of Power Surveillance Requirements.							
7. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	(6)	(6)	(8)	1, 2, 3, 4
b. RWST Level--Low-Low				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Coincident With: Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Loss of Power								
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	N.A.		N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	N.A.		N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	N.A.		N.A.		N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.			N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.			N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	1, 2, 3
10. Control Room Ventilation								
a. Manual Initiation	N.A.	N.A.	N.A.		N.A.	N.A.	N.A.	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK(9)	CHANNEL CALIBRATION(9)	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST(7)(9)	TRIP ACTUATING DEVICE OPERATIONAL TEST(9)	ACTUATION LOGIC TEST(9)	MASTER RELAY TEST(9)	SLAVE RELAY TEST(9)	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Control Room Ventilation (Continued)								
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	(6)	N.A.	N.A.	1, 2, 3, 4
d. Control Room Intake Air Radioactivity-High				N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
e. Loss of Power	See Item 8. above for all Loss of Power Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(1) Each train shall be tested at the frequency specified in the Surveillance Frequency Control Program.

(2) Deleted

(3) Deleted

(4) Deleted

(5) Deleted

(6) Each actuation train shall be tested at a frequency in accordance with the Surveillance Frequency Control Program. Testing of each actuation train shall include master relay testing of both logic trains. If an ESFAS instrumentation channel is inoperable due to failure of the Actuation Logic Test and/or Master Relay Test, increase the surveillance frequency such that each train is tested at the frequency specified in the Surveillance Frequency Control Program unless the failure can be determined by performance of an engineering evaluation to be a single random failure.

(7) For channels with bypass test instrumentation, input relays are tested at a frequency in accordance with the Surveillance Frequency Control Program.

(8) The test interval is R for Potter & Brumfield MDR Series slave relays.

(9) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

3/4.3.3 (Not Used)

TABLE 3.3-6

(Not Used)

PAGE INTENTIONALLY BLANK

TABLE 4.3-3 NOT USED

3.3.3.2 through 3.3.3.4 (These specifications are not used)

Pages 3/4 3-55 through 3/4 3-60 have been deleted.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown System Functions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one or more required channels of one or more Remote Shutdown System Functions inoperable, restore the inoperable Function(s) to OPERABLE status within 30 days, or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

NOTE: Separate condition entry is allowed for each Function.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each normally energized Remote Shutdown System monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at a frequency in accordance with the Surveillance Frequency Control Program.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit including the actuated components, shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program.

4.3.3.5.3 Each Remote Shutdown System required instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CALIBRATION at a frequency in accordance with the Surveillance Frequency Control Program. [NOTE: Neutron detectors and reactor trip breaker indication are excluded from CHANNEL CALIBRATION.]

NOTE: This corrected page was issued by letter dated February 8, 2005

TABLE 3.3-9 (NOT USED)

TABLE 3.3-9 (NOT USED)

TABLE 3.3-9 (NOT USED)

TABLE 3.3-9 (NOT USED)

TABLE 4.3-6 (NOT USED)

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

As shown in Table 3.3-10.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure	4	1	43
2. Reactor Coolant Outlet Temperature- T_{HOT} (Wide Range)	4 (1/loop)	4	35
3. Reactor Coolant Inlet Temperature- T_{COLD} (Wide Range)	4 (1/loop)	4	35
4. Reactor Coolant Pressure - Wide Range and Extended Range	3	1	37
5. Pressurizer Water Level	4	1	43
6. Steam Line Pressure	4/steam generator	1/steam generator	43
7. Steam Generator Water Level - Narrow Range	4/steam generator	1/steam generator	43
8. Steam Generator Water Level - Wide Range	4 (1/steam generator)	4	35
9. Refueling Water Storage Tank Water Level	3	1	37
10. Auxiliary Feedwater Storage Tank Water Level	3	1	37
11. Auxiliary Feedwater Flow	4 (1/steam generator)	4	35
12. Reactor Coolant System Subcooling Margin Monitoring	2	1	36

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
13. Containment Water Level (Narrow Range)	2	1	36
14. Containment Water Level (Wide Range)	3	1	37
15. Core Exit Thermocouples	**2	**1	42
16. Steam Line Radiation Monitor	1/steam line	1/steam line	40
17. Containment - High Range Radiation Monitor	2	1	39
18. Reactor Vessel Water Level (RVWL)	2*	1*	41
19. Neutron Flux (Extended Range)	2	1	42
20. Not Used			
21. Containment Pressure (Extended Range)	2	1	36
22. Steam Generator Blowdown Radiation Monitor	1/blowdown line	1/blowdown line	40
23. Neutron Flux - Startup Rate (Extended Range)	2	1	42

* A channel is eight sensors in a probe. A channel is OPERABLE if four or more sensors, one or more in the upper section section and three or more in the lower section, are OPERABLE.

** A channel is OPERABLE if at least two core exit thermocouples per core quadrant are OPERABLE, and at least one quadrant has at least four OPERABLE thermocouples.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 35 - a. With the number of OPERABLE channels one less than the Minimum Channels Operable requirement, restore the inoperable channel to OPERABLE status within 30 days or submit a Special Report within the next 14 days describing 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.

b. With the number of OPERABLE channels two or more less than the Minimum Channels Operable requirement, restore one or more of the inoperable channels such that at least three channels are in OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 36 - a. With the number of OPERABLE channels one less than the Total Number of Channels requirements, restore one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less than the Minimum Channels Operable requirements, restore at least one inoperable channel to OPERABLE status within 48 hours, or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 37 - a. With the number of OPERABLE channels one less than the Total Number of Channels requirements, restore the inoperable channel to OPERABLE status within 31 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels two less than the Total Number of Channels requirement, restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels less than the Minimum Channels Operable requirement, restore at least one inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS (Continued)

ACTION 39 - a. With the number of OPERABLE channels one less than the Total Number of Channels requirements, restore one inoperable channel to OPERABLE status within 30 days, or submit a Special Report within the next 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the inoperable instrumentation channels to OPERABLE status.

b. With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, restore at least one inoperable channel to OPERABLE status within 7 days, or submit a Special Report within the next 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the inoperable instrumentation channels to OPERABLE status.

ACTION 40 - a. With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements and with a functional diverse channel, restore at least one inoperable channel to OPERABLE status within 30 days, or submit a Special Report within the next 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the inoperable instrumentation channels to OPERABLE status.

b. With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements and with the diverse channel not functional, restore at least one inoperable channel to OPERABLE status within 7 days or submit a Special Report within the next 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the inoperable instrumentation channels to OPERABLE status.

ACTION 41 - a. With the number of OPERABLE channels one less than the Required Number of Channels, either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or submit a Special Report 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.

b. With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;
2. Submit a Special Report within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and
3. Restore the system to OPERABLE status at the next scheduled refueling.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS (Continued)

ACTION 42 - a. With one required channel inoperable, restore the required channel to OPERABLE status within 30 days; otherwise, a Special Report shall be submitted within the next 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 to OPERABLE status.

b. With two required channels inoperable, restore one required channel to OPERABLE status within 7 days; otherwise, be in HOT STANDBY within 6 hours, and in HOT SHUTDOWN in the next 6 hours.

ACTION 43 - a. With the number of OPERABLE channels two less than the Total Number of Channels requirements, restore the inoperable channel to OPERABLE status within 31 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels three less than the Total Number of Channels requirement, restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels less than the Minimum Channels Operable requirement, restore at least one inoperable channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.3-7 has been deleted.

Page 3/4 3-74 has been deleted.

3.3.3.7 through 3.3.3.11 and 3.3.4 (These specification numbers are not used)

Pages 3/4 3-76 through 3/4 3-84 have been deleted.

3/4.3.5 ATMOSPHERIC STEAM RELIEF VALVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION:

3.3.5.1 The atmospheric steam relief valve instrumentation shown in Table 3.3-14 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-14

ACTION: As shown in Table 3.3-14

SURVEILLANCE REQUIREMENTS:

4.3.5.1 Perform a CHANNEL CHECK on each atmospheric steam relief valve automatic actuation channel at a frequency in accordance with the Surveillance Frequency Control Program.

4.3.5.2 Perform a CHANNEL CALIBRATION on each atmospheric steam relief valve automatic actuation channel at a nominal setpoint of 1225 psig ± 7 psi at a frequency in accordance with the Surveillance Frequency Control Program.

4.3.5.3 Perform an ANALOG CHANNEL OPERATIONAL TEST on each atmospheric steam relief valve automatic actuation channel at a nominal setpoint of 1225 psig ± 7 psi at a frequency in accordance with the Surveillance Frequency Control Program.

TABLE 3.3-14

ATMOSPHERIC STEAM RELIEF VALVE INSTRUMENTATION

FUNCTIONAL UNIT	REQUIRED NO. OF CHANNELS	APPLICABLE MODES	ACTION
Manual actuation control channels	4 (1 / valve)	1, 2,3,4*	1
Automatic actuation control channels	4 (1 / valve)	1,2#	2

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TABLE 3.3-14 (Continued)

TABLE NOTATIONS

* When steam generators are being used for decay heat removal.

Atmospheric steam relief valve(s) may be in manual operation and open, or in automatic operation, to maintain secondary side pressure at or below an indicated steam generator pressure of 1225 psig.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the required number of channels, declare the affected valve(s) inoperable and apply Technical Specification 3.7.1.6.

ACTION 2 - a. With one less than the required number of OPERABLE channels, within 7 days restore the inoperable channel to OPERABLE status or apply the requirements of the CRMP; or be in at least HOT STANDBY within the next 6 hours.

b. With two less than the required number of OPERABLE channels, within 72 hours restore at least three channels to OPERABLE status or apply the requirements of the CRMP; or be in at least HOT STANDBY within the next 6 hours.

c. With more than two less than the required number of OPERABLE channels, within 1 hour restore at least two channels to OPERABLE status or apply the requirements of the CRMP; or be in at least HOT STANDBY within the next 6 hours.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions Specification 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE and with two reactor coolant loops in operation when the Reactor Trip System breakers are closed and one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

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

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

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

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

APPLICABILITY: MODE 3.

ACTION:

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

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN of LCO 3.1.1 and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

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

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% narrow range at a frequency in accordance with the Surveillance Frequency Control Program.

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

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet SHUTDOWN MARGIN of LCO 3.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

e. RHR Loop A with valve CV0198 locked or pinned in position to limit flow to 125 gpm,

f. RHR Loop B with valve CV0198 locked or pinned in position to limit flow to 125 gpm, and

g. RHR Loop C with valve CV0198 locked or pinned in position to limit flow to 125 gpm.

APPLICABILITY: MODE 4.

ACTION:

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

b. With no loop in operation, suspend operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN of LCO 3.1.1 and immediately initiate corrective action to return the required loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet SHUTDOWN MARGIN of LCO 3.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% narrow range at a frequency in accordance with the Surveillance Frequency Control Program.

4.4.1.3.3 At least one reactor coolant loop, or one RHR loop with valve CV0198 locked or pinned in position to limit flow to 125 gpm shall be verified in operation and circulating reactor coolant at a frequency in accordance with the Surveillance Frequency Control Program.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop with valve CV0198 locked or pinned in position to limit flow to 125 gpm shall be OPERABLE and in operation*, and either:

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

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

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

ACTION:

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

b. With no RHR loop in operation, suspend operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN of LCO 3.1.1 and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.1.2 At least one RHR loop with valve CV0198 locked or pinned in position to limit flow to 125 gpm shall be determined to be in operation and circulating reactor coolant at a frequency in accordance with the Surveillance Frequency Control Program.

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* The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet SHUTDOWN MARGIN of LCO 3.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2

a. At least two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation**, and

b. Each valve or mechanical joint used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

b. With no RHR loop in operation, suspend all operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN of LCO 3.1.1 and immediately initiate corrective action to return the required RHR loop to operation.

c. With a valve or mechanical joint used to isolate unborated water sources not secured in the closed position, immediately suspend all operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet SHUTDOWN MARGIN specified in the Core Operating Limits Report (COLR) and initiate action to secure the valve(s) or joint(s) in the closed position and within 4 hours verify the SHUTDOWN MARGIN is within limits specified in the COLR. The required action to verify the SHUTDOWN MARGIN within limits must be completed whenever ACTION c is entered. A separate ACTION entry is allowed for each unsecured valve or mechanical joint.

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.2.2 Each valve or mechanical joint used to isolate unborated water sources shall be verified closed and secured in position at a frequency in accordance with the Surveillance Frequency Control Program.

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

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause introduction into the RCS of coolant with boron concentration less than that required to meet SHUTDOWN MARGIN of LCO 3.1.1, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

Page 3/4 4-7 has been deleted.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting[1] of 2485 psig +2%, -3%.[2]

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

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

[2]The as left lift setting shall be within ±1% following valve testing.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1816 cubic feet, and at least two groups of pressurizer heaters supplied by ESF power each having a capacity of at least 175 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters supplied by ESF power shall be verified by energizing the heaters and measuring circuit current at a frequency in accordance with the Surveillance Frequency Control Program.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

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

d. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV in closed position; within 72 hours restore the block valve to OPERABLE status or apply the requirements of the CRMP; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With both block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place the associated PORVs in the closed position; restore at least one block valve to OPERABLE status within the next hour or apply the requirements of the CRMP; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by:

a. Performing a CHANNEL CALIBRATION on the actuation channel, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed in accordance with the ACTIONS of Specification 3.4.4.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

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

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

ACTION:

NOTE: Separate entry is allowed for each steam generator tube

a. With one or more steam generator tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program, within 7 days verify tube integrity of the affected tube(s) is maintained until the next inspection, or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

   AND

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

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

SURVELLANCE REQUIREMENTS

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

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

PAGES 3/4 4-14 THROUGH 3/4 4-18 HAVE BEEN DELETED.

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 Atmosphere Radioactivity Monitor (particulate channel), and

b. The Containment Normal Sump Level and Flow Monitoring System.

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

ACTION:

a. With the required containment atmosphere radioactivity monitor inoperable perform the following actions or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:

   1) Restore the containment atmosphere monitor (particulate channel) to OPERABLE status within 30 days and,

   2) Obtain and analyze a grab sample of the containment atmosphere for particulate radioactivity at least once per 24 hours, or

   3) Perform a Reactor Coolant System water inventory balance at least once per 24 hours.

b. With the required containment normal sump level and flow monitoring system inoperable perform the following actions or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:

   1) Restore the containment normal sump and flow monitoring system to OPERABLE status within 30 days and,

   2) Perform a Reactor Coolant System water inventory balance at least once per 24 hours.

c. With both a. and b. inoperable, enter 3.0.3.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Monitoring (particulate channel) performance of the following:

   1) CHANNEL CHECK at a frequency in accordance with the Surveillance Frequency Control Program, and
   2) CHANNEL CALIBRATION at a frequency in accordance with the Surveillance Frequency Control Program

b. Containment Normal Sump Level and Flow Monitoring System performance of CHANNEL CALIBRATION at a frequency in accordance with the Surveillance Frequency Control Program.

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 of primary-to-secondary leakage through any one steam generator,

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

e. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, isolate the affected component, pipe, or vessel from the RCS 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 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With Reactor Coolant System operational UNIDENTIFIED or IDENTIFIED LEAKAGE greater than the above limits, reduce leakage to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With primary-to-secondary leakage not within the limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.1 Note: this requirement is not applicable to primary-to-secondary leakage (refer to 4.4.6.2.3).

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

a. Monitoring the containment atmosphere particulate radioactivity channel at a frequency in accordance with the Surveillance Frequency Control Program;

b. Monitoring the containment normal sump inventory and discharge at a frequency in accordance with the Surveillance Frequency Control Program;

c. Performing a Reactor Coolant System water inventory balance at a frequency in accordance with the Surveillance Frequency Control Program; and [1]

d. Monitoring the Reactor Head Flange Leakoff System at a frequency in accordance with the Surveillance Frequency Control Program.

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

a. At a frequency in accordance with the Surveillance Frequency Control Program,

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,

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

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve except for valves XRH0060 A, B, C, and XRH0061 A, B, C.

4.4.6.2.3 Primary-to-secondary leakage shall be verified ≤ 150 gallons per day through any one steam generator at a frequency in accordance with the Surveillance Frequency Control Program. [1]

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
XSI0007 A, B, C	HHSI Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0009 A, B, C	HHSI Hot Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XSI0010 A, B, C	LHSI/HHSI Hot Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XRH0020 A, B, C	LHSI Hot Leg Recirculation Check Valves (RCS Loops 1, 2, 3)
XRH0032 A, B, C	LHSI/RHR Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0038 A, B, C	LHSI/HHSI/RHR/Accumulator Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XSI0046 A, B, C	Accumulator Cold Leg Injection Check Valves (RCS Loops 1, 2, 3)
XRH0060 A, B, C	RHR Suction Isolation Valves (RCS Loops 1, 2, 3)
XRH0061 A, B, C	RHR Suction Isolation Valves (RCS Loops 1, 2, 3)

3/4.4.7 (This specification not used)

Pages 3/4 4-24 and 3/4 4-25 have been deleted

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

   1. Verify DOSE EQUIVALENT I-131 ≤ 60 microCuries per gram once every 4 hours, and

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

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

c. With the requirements of ACTION a or ACTION b not met or DOSE EQUIVALENT I-131 exceeding 60 microCuries per gram, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours.

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

SURVEILLANCE REQUIREMENTS

4.4.8.1 Verify Reactor Coolant System DOSE EQUIVALENT XE-133 specific activity ≤ 540 microCuries per gram at a frequency in accordance with the surveillance frequency control program.

4.4.8.2 Verify Reactor Coolant System DOSE EQUIVALENT I-131 specific activity ≤ 1.0 microCuries per gram:

   a. At a frequency in accordance with the surveillance frequency control program, and

   b. Between 2 and 6 hours after THERMAL POWER change of 15 % or greater RATED THERMAL POWER within a 1 hour period.

(This page not used)

FIGURE 3.4-1

(This figure not used)

Table 4.4-4

(This table not used)

(This page not used)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at a frequency 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, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL - RV INTERMEDIATE SHELL R-1606-3
COPPER CONTENT: CONSERVATIVELY ASSUMED AS 0.10 WT%

RT_{NDT} INITIAL: 10°F
RT_{NDT} AFTER 32 EFPY
1/4, 91°F
3/4T, 64°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 100°/HR FOR THE SERVICE PERIOD UP TO 32 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

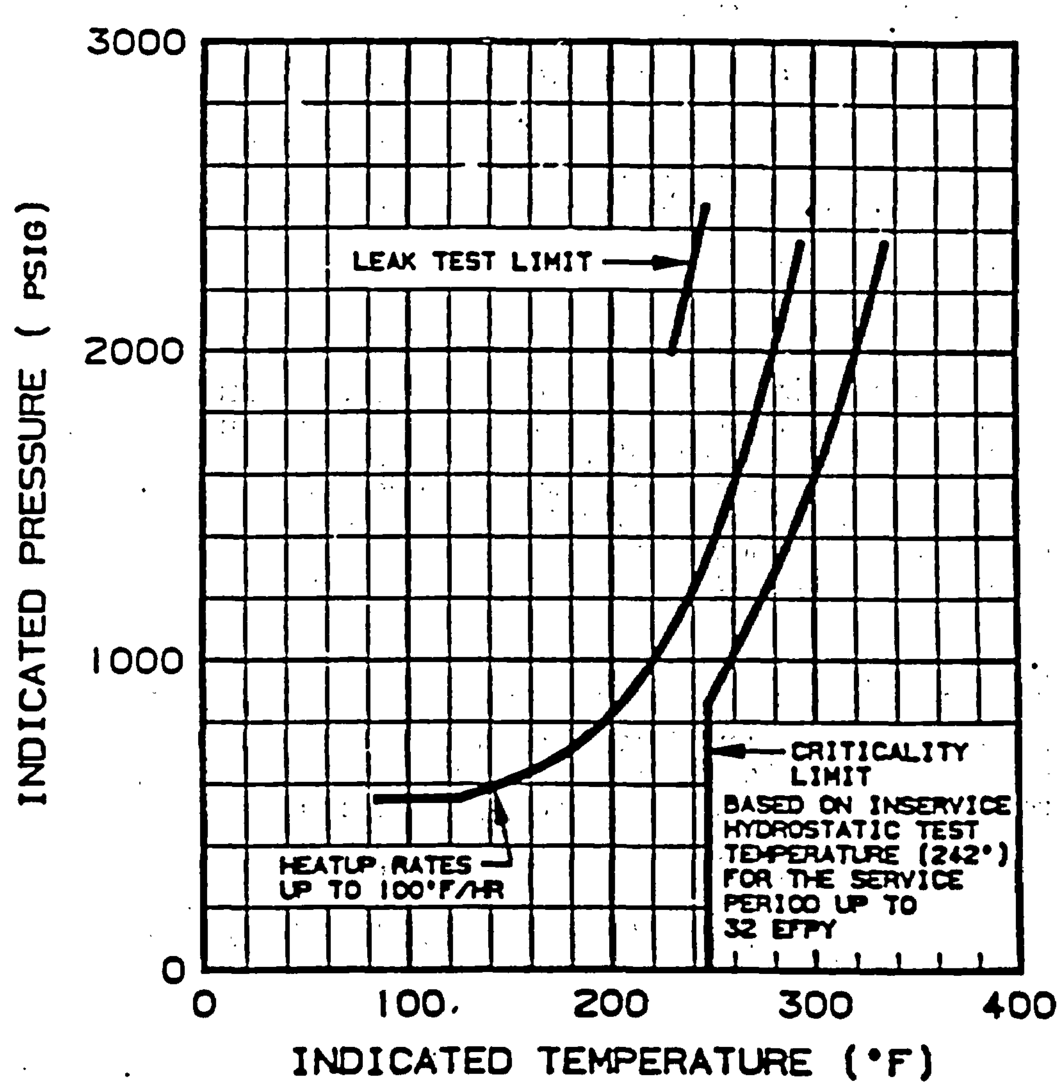

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 32 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL - RV
INTERMEDIATE SHELL R-1606-3
COPPER CONTENTS: CONSERVATIVELY
ASSUMED AS 0.10 WT%

RT_{NDT} INITIAL: 10°F
RT_{NDT} AFTER 32 EFPY
1/4, 91°F
3/4T, 64°F

SINGLE CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°/HR FOR THE SERVICE PERIOD UP TO 32 EFPY, AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

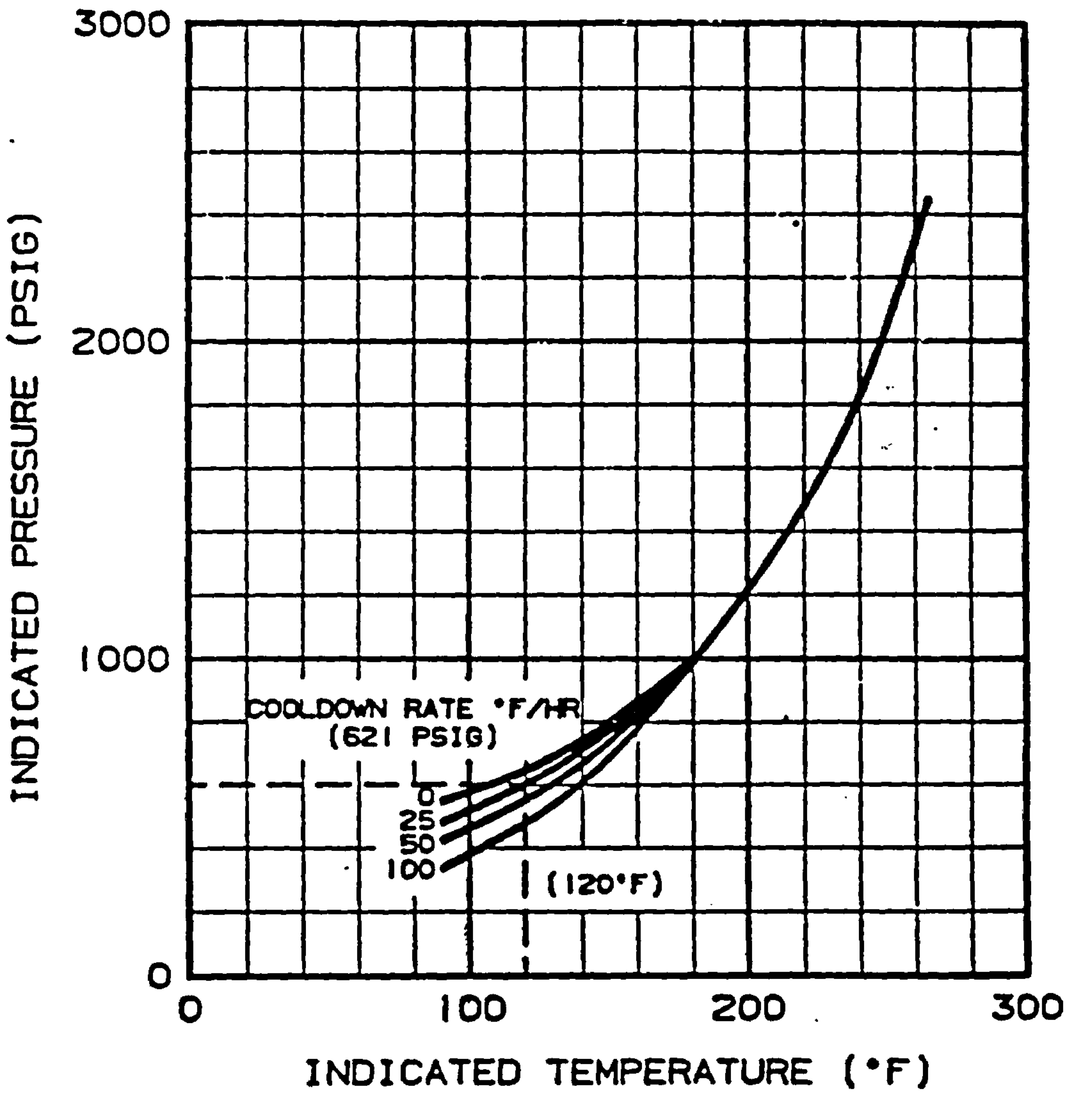

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 32 EFPY

TABLE 4.4-5

(This table number not used)

3.4.9.2 (This specification not used)

LIMITING CONDITION FOR OPERATION

3.4.9.3 An Overpressure Protection System shall be OPERABLE with a maximum of one centrifugal charging pump capable of injecting into the RCS and the emergency core cooling system (ECCS) accumulators isolated and either a. or b. below:

a. Two power-operated relief valves (PORVs) with lift settings which do not exceed the limit established in Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.0 square inches.

APPLICABILITY: MODES 4 and 5, and MODE 6 when the head is on the reactor vessel[1, 5]

ACTION:

a. With one or more ECCS accumulators not isolated, isolate the ECCS accumulator(s) within 1 hour.

b. With more than one centrifugal charging pump capable of injecting into the RCS, immediately initiate action to render all but one centrifugal charging pump incapable of injecting into the RCS[2].

c. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.0 square inch vent within the next 8 hours.

d. With one PORV inoperable in MODES 5 or 6 with the head on the reactor vessel, restore the inoperable PORV to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 2.0 square inch vent within the next 8 hours[3].

e. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.0 square inch vent within 8 hours[3].

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

g. The provisions of Specification 3.0.4.b are not applicable when entering MODE 4.

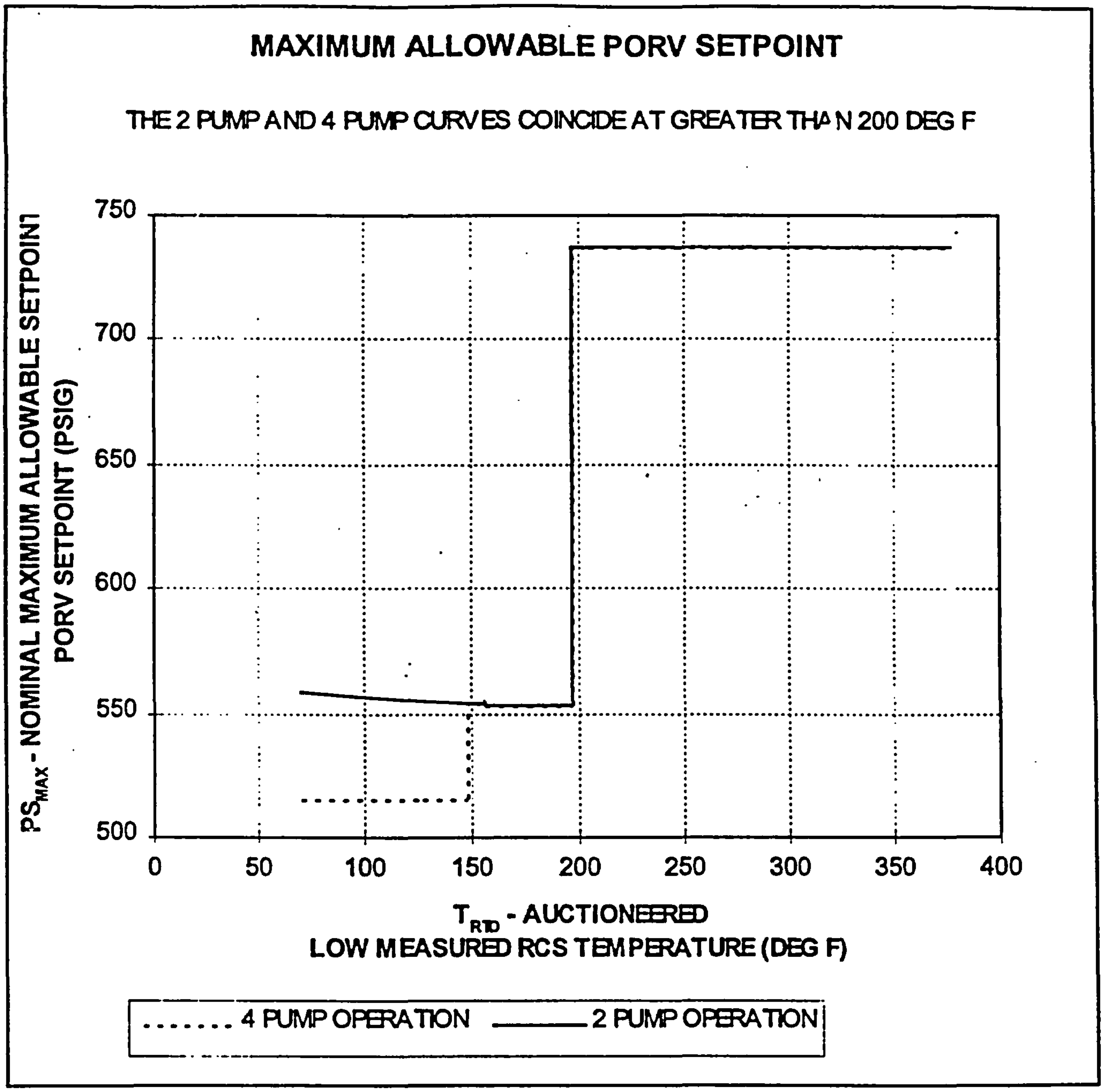

FIGURE 3.4-4

NOMINAL MAXIMUM ALLOWABLE PORV
SETPOINT FOR THE COLD OVERPRESSURE SYSTEM

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at a frequency in accordance with the Surveillance Frequency Control Program thereafter when the PORV is required OPERABLE;

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

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

4.4.9.3.2 The RCS vent(s) shall be verified to be open at a frequency in accordance with the Surveillance Frequency Control Program[4] when the vent(s) is being used for overpressure protection.

4.4.9.3.3 The positive displacement pump shall be demonstrated inoperable[5] at a frequency in accordance with the Surveillance Frequency Control Program, except when the reactor vessel head is removed or when both centrifugal charging pumps are inoperable and secured, by verifying that the motor circuit breakers are secured in the open position.[2]

4.4.9.3.4 Verify at a frequency in accordance with the Surveillance Frequency Control Program that only one centrifugal charging pump is capable of injecting into the RCS[5], except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.[2]

4.4.9.3.5 Verify at a frequency in accordance with the Surveillance Frequency Control Program that each ECCS accumulator is isolated.

SPECIFICATION NOTATIONS

[1] ECCS accumulator isolation is required only when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by Figures 3.4-2 and 3.4-3.

[2] An inoperable centrifugal charging pump(s) and/or positive displacement charging pump may be energized for testing or pump switching provided the discharge of the pump(s) has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position. Reactor coolant pump seal injection flow may be maintained during the RCS isolation process.

[3] This ACTION may be suspended for up to 7 days to allow functional testing to verify PORV operability. During this test period, operation of systems or components which could result in an RCS mass or temperature increase will be administratively controlled. During the ASME stroke testing of two inoperable PORVS, cold overpressurization mitigation will be provided by two RHR discharge relief valves associated with two OPERABLE and operating RHR loops which have the auto closure interlock bypassed [or deleted]. If one PORV is inoperable, cold overpressure mitigation will be provided by the OPERABLE PORV and one RHR discharge relief valve associated with an OPERABLE and operating RHR loop which has the auto closure interlock bypassed [or deleted].

[4] Except when the vent pathway is provided with a valve that is locked, sealed, or otherwise secured in the open position, then verify these valves open at a frequency in accordance with the Surveillance Frequency Control Program.

[5] Entry into MODE 4 from MODE 3 is permitted while making all but one centrifugal charging pump incapable of injecting into the RCS pursuant to Specification 4.4.9.3.4, and for the positive displacement pump declared inoperable pursuant to Specification 4.4.9.3.3 provided that all but one centrifugal charging pump is made incapable of injecting into the RCS, and the positive displacement pump is declared inoperable within 4 hours after entry into MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be ultrasonically examined over the volume from the inner bore of the flywheel to the circle of one-half the outer radius at a frequency in accordance with the Surveillance Frequency Control Program and shall comply with regulatory positions C.4.b (3), (4), and (5) of Regulatory Guide 1.14, Revision 1, August 1975.

3.4.11 (This specification not used)

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Safety Injection System accumulator shall be OPERABLE

APPLICABILITY: MODES 1 and 2
MODE 3 with pressurizer pressure > 1000 psig

ACTION:

a. With one accumulator inoperable, except as a result of boron concentration outside the required limits, within 24 hours restore the inoperable accumulator to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With more than one accumulator inoperable, except as a result boron concentration outside the required limits, within 1 hour restore at least two accumulators to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With the boron concentration of one accumulator outside the required limit, within 72 hours restore the boron concentration to within the required limits or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

d. With the boron concentrations of more than one accumulator outside the required limit, within 1 hour restore the boron concentration of at least two accumulators to within the required limits or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying the contained borated water volume is ≥ 8800 gallons and ≤ 9100 gallons and nitrogen cover-pressure is ≥ 590 psig[1] and ≤ 670 psig, and

2) Verifying that each accumulator isolation valve is open.

b. At a frequency in accordance with the Surveillance Frequency Control Program and within 6 hours* after each solution volume increase of greater than or equal to 1% of tank volume that is not the result of addition from the RWST by verifying the boron concentration of the accumulator solution is ≥ 2700 ppm and ≤ 3000 ppm, and

c. At a frequency in accordance with the Surveillance Frequency Control Program when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is removed.

* The 6 hr. SR is only required to be performed for affected accumulators.
[1] For Unit 1 only, the nitrogen cover-pressure is verified to be ≥ 500 psig for the remainder of Unit 1 Cycle 23.

NOT USED

EMERGENCY CORE COOLING SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE High Head Safety Injection pump,

b. One OPERABLE Low Head Safety Injection pump,

c. One OPERABLE RHR heat exchanger, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation through a High Head Safety Injection pump and into the Reactor Coolant System and through a Low Head Safety Injection pump and its respective RHR heat exchanger into the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With less than the above subsystems OPERABLE, but with at least two High Head Safety Injection pumps in an OPERABLE status, two Low Head Safety Injection pumps and associated RHR heat exchangers in an OPERABLE status, and sufficient flow paths to accommodate these OPERABLE Safety Injection pumps and RHR heat exchangers,** within 7 days restore the inoperable subsystem(s) to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With less than two of the required subsystems OPERABLE, within 1 hour restore at least two subsystems to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

---

* Entry into MODE 3 is permitted for the Safety Injection pumps declared inoperable pursuant to Specification 4.5.3.1.2 provided that the Safety Injection pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

** Verify required pumps, heat exchangers and flow paths OPERABLE every 48 hours.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

c. With less than the required flow paths OPERABLE solely due to potential effects of LOCA generated and transported debris that exceeds analyzed amounts, perform the following:

1. immediately initiate action to implement compensatory actions,

AND

2. within 90 days restore the affected flowpath(s) to OPERABLE status,

OR

Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN 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 submitted within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

Valve Number		Valve Function	Valve Position
XSI0008	A, B, C	High Head Hot Leg Recirculation Isolation	Closed
XRH0019	A, B, C	Low Head Hot Leg Recirculation Isolation	Closed

b. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

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

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

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

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At a frequency in accordance with the Surveillance Frequency Control Program by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components show no evidence of structural distress or abnormal corrosion.

SURVEILLANCE REQUIREMENTS (Continued)

e. At a frequency 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 an Automatic Switchover to Containment Sump test signal, and

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

  a) High Head Safety Injection pump, and

  b) Low Head Safety Injection pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

 1) High Head Safety Injection pump $\geq$ 1480 psid, and

 2) Low Head Safety Injection pump $\geq$ 286 psid.

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

 1) For High Head Safety Injection pump lines, with the High Head Safety Injection pump running, the pump flow rate is greater than 1470 gpm and less than 1620 gpm.

 2) For Low Head Safety Injection pump lines, with the Low Head Safety Injection pump running, the pump flow rate is greater than 2550 gpm and less than 2800 gpm.

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

LIMITING CONDITION FOR OPERATION

3.5.3.1 As a minimum, the following ECCS components shall be OPERABLE:

a. Two OPERABLE High Head Safety Injection pumps,*

b. Two OPERABLE Low Head Safety Injection pumps and their associated RHR heat exchangers, and

c. Two OPERABLE flow paths capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation through a High Head Safety Injection pump and into the Reactor Coolant System and through a Low Head Safety Injection pump and its respective RHR heat exchanger into the Reactor Coolant System.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above-required ECCS components OPERABLE because of the inoperability of either the High Head Safety Injection pumps or the flow paths from the refueling water storage tank, restore at least the required ECCS components to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With less than the above-required ECCS components OPERABLE because of the inoperability of either the residual heat removal heat exchangers or the Low Head Safety Injection pumps, restore at least the required ECCS components to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

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

d. Specification 3.0.4.b is not applicable to the High Head Safety Injection pumps.

*A maximum of one High Head Safety Injection pump shall be OPERABLE and a second High Head Safety Injection pump shall be OPERABLE except that its breaker shall be racked out (the third HHSI pump shall have its breaker racked out) within: (1) 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever comes first; or (2) 4 hours after entering MODE 4 from MODE 5 or prior to the temperature of one or more of the RCS cold legs exceeding 225°F, whichever comes first.

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

SURVEILLANCE REQUIREMENTS

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

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

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

EMERGENCY CORE COOLING SYSTEMS

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

LIMITING CONDITION FOR OPERATION

3.5.3.2 All High Head Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all High Head Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

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

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

3/4.5.4 (This specification number is not used.)

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 458,000 gallons, and

b. A boron concentration between 2800 ppm and 3000 ppm.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by:

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

b. Verifying the boron concentration of the water.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.6 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.6 Three independent Residual Heat Removal (RHR) loops shall be OPERABLE with each loop comprised of:

a. One OPERABLE RHR pump,

b. One OPERABLE RHR heat exchanger, and

c. One OPERABLE flowpath capable of taking suction from its associated RCS hot leg and discharging to its associated RCS cold leg.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one RHR loop inoperable, within 7 days restore the required loop to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two RHR loops inoperable, within 24 hours restore at least two RHR loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three RHR loops inoperable, immediately initiate corrective action to restore at least one RHR loop to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.5.6.1 Each RHR loop shall be demonstrated OPERABLE on a STAGGERED TEST BASIS pursuant to the requirements of Specification 4.0.5.

4.5.6.2 At a frequency in accordance with the Surveillance Frequency Control Program by verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that:

a. With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 350 psig, the interlocks prevent the valves from being opened.

---

*Valves MOV-0060 A, B, and C and MOV-0061 A, B, and C may have power removed to support the FHAR (Fire Hazard Analysis Report) assumptions.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

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

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than at a frequency in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With either the measured overall integrated containment leakage rate or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding the allowances in the Containment Leakage Rate Testing Program, restore the overall integrated leakage rate and the combined leakage rate for all penetrations subject to Type B and C tests to within the allowances in the Containment Leakage Rate Testing Program within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.2 are not applicable.

Pages 3/4 6-3 and 3/4 6-4 have been deleted.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

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

ACTION:

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

a. With only one containment air lock door inoperable:

1. Verify the OPERABLE air lock door is closed within 1 hour and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

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

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

b. With only the containment air lock interlock mechanism inoperable:

1. Verify an OPERABLE air lock door is closed within 1 hour and lock an OPERABLE air lock door closed within 24 hours;

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

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (Entry into and exit from containment is permissible under the control of a dedicated individual); and

c. With the containment air lock inoperable, except as specified in ACTION a. or ACTION b. above, immediately initiate action to evaluate overall containment leakage rate per Specification 3.6.1.2 and verify an air lock door is closed within 1 hour. Restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program.

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

c. By verifying at a frequency in accordance with the Surveillance Frequency Control Program that the instrument air pressure in the header to the personnel airlock seals is $\geq$ 90 psig.

d. By verifying the door seal pneumatic system OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by conducting a seal pneumatic system leak test and verifying one of the following:

1) That system pressure does not decay more than 1.5 psi from 90 psig minimum within 24 hours, or

2) That system pressure does not decay more than .50 psi from 90 psig minimum within 8 hours.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.1 and +0.3 psig.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 110°F.

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

ACTION:

With the containment average air temperature greater than 110°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of a minimum of four RCFC inlet temperatures and shall be determined at a frequency in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment(s) shall be maintained as required by the Containment Post-Tensioning System Surveillance Program.

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

ACTION:

If the containment is not OPERABLE, restore containment to OPERABLE status in 1 hour, or be in at least HOT STANDBY in the next 6 hours and be in COLD SHUTDOWN in the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 CONTAINMENT PRESTRESSING SYSTEM

Verify containment structural integrity in accordance with the Containment Post-Tensioning System Surveillance Program. The provisions of SR 4.0.2 do not apply to extending the interval for this surveillance.

PAGES 3/4 6-10, 6-11, 6-11A, AND 6-11B HAVE BEEN DELETED.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 48-inch containment shutdown purge supply and exhaust isolation valve shall be closed and sealed closed, and

b. The 18-inch supplementary containment purge supply and exhaust isolation valves shall be closed to the maximum extent practicable but may be open for supplementary purge system operation for pressure control, for ALARA and respirable air quality considerations for personnel entry and for surveillance tests that require the valves to be open.

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

ACTION:

a. With a 48-inch containment purge supply and/or exhaust isolation valve open or not sealed closed, close and/or seal close that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 18-inch supplementary containment purge supply and/or exhaust isolation valve(s) open for reasons other than given in Specification 3.6.1.7.b. above, close the open 18-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.2 and/or 4.6.1.7.3, restore the inoperable valve(s) to OPERABLE status or isolate the penetrations so that the measured leakage rate does not exceed the limits of Specifications 4.6.1.7.2 and/or 4.6.1.7.3 within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 48-inch containment purge supply and exhaust isolation valve shall be verified to be sealed closed at a frequency in accordance with the Surveillance Frequency Control Program.

4.6.1.7.2 At a frequency in accordance with the Surveillance Frequency Control Program, the inboard and outboard isolation valves with resilient material seals in each sealed closed 48-inch containment purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.

4.6.1.7.3 At a frequency in accordance with the Surveillance Frequency Control Program each 18-inch supplementary containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.01 L_a when pressurized to P_a.

4.6.1.7.4 At a frequency in accordance with the Surveillance Frequency Control Program each 18-inch supplementary containment purge supply and exhaust isolation valve shall be verified to be closed or open in accordance with Specification 3.6.1.7.b.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Three independent Containment Spray Systems shall be OPERABLE with each Spray system capable of taking suction from the RWST and transferring suction to the containment sump.

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

ACTION:

a. With one Containment Spray System inoperable, within 7 days restore the inoperable Spray System to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

b. With more than one Containment Spray System inoperable, within 1 hour restore at least two Spray Systems to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more Containment Spray Systems inoperable in MODE 1, 2, or 3 solely due to potential effects of LOCA generated and transported debris that exceeds analyzed amounts, perform the following:

   1. immediately initiate action to implement compensatory actions,

   AND

   2. within 90 days restore the affected system(s) to OPERABLE status,

   OR

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

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

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

b. By verifying on a STAGGERED TEST BASIS, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 283 psid when tested pursuant to Specification 4.0.5;

c. At a frequency in accordance with the Surveillance Frequency Control Program during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure High 3 test signal, and

   2) Verifying that each spray pump starts automatically on a Containment Pressure High 3 test signal coincident with a sequencer start signal.

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

RECIRCULATION FLUID PH CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The recirculation fluid pH control system shall be operable with between 11,500 lbs. (213 cu. ft.) and 15,100 lbs (252 cu. ft.) of trisodium phosphate (w/12 hydrates) available in the storage baskets in the containment.

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

ACTION:

With the amount of trisodium phosphate outside the specified range, restore the system to the correct amount within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the system to the correct amount within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 At a frequency in accordance with the Surveillance Frequency Control Program, the recirculation fluid pH control system shall be demonstrated operable by visually verifying that:

a. 6 trisodium phosphate storage baskets are in place, and

b. have maintained their integrity, and

c. are filled with trisodium phosphate such that the level is within the specified range.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Three independent groups of Reactor Containment Fan Coolers (RCFC) shall be OPERABLE with a minimum of two units in two groups and one unit in the third group.

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

ACTION:

A. With one group of the above required Reactor Containment Fan Coolers inoperable, within 7 days restore the inoperable group of RCFC to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With more than one group of the above required Reactor Containment Fan Coolers inoperable, within 1 hour restore at least two groups of RCFC to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of Reactor Containment Fan Coolers shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Starting each non-operating fan group from the control room, and verifying that each fan group operates for at least 15 minutes, and

2) Verifying a component cooling water flow rate of greater than or equal to 1800 gpm to each cooler.

b. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that each fan group starts automatically on a Safety Injection test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE with isolation times less than or equal to the required isolation times.

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation barrier[(2)] OPERABLE in each affected penetration that is open and within 24 hours:

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 check valve with flow through the valve secured[(1)] [(3)], or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange[(1)], or

d. Apply the requirements of the CRMP.

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

SURVEILLANCE REQUIREMENTS

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

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at a frequency in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Containment Ventilation Isolation test signal, each purge and exhaust valve actuates to its isolation position; and

c. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position.

d. Verifying that on a Phase "A" Isolation test signal, coincident with a low charging header pressure signal, that each seal injection valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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(1) Penetration flow paths (except for Containment Purge flow paths) may be unisolated intermittently under administrative controls.
(2) An isolation barrier may either be a closed system (i.e., General Design Criteria 57 penetrations) or an isolation valve.
(3) A check valve may not be used to isolate an affected penetration flow path in which more than one isolation valve is inoperable or in which the isolation barrier is a closed system with a single isolation valve (i.e., General Design Criteria 57 penetration)

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Section 3.6.4.1 has been deleted

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Section 3.6.4.2 has been deleted

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided that within 24 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 There are no additional requirements other than those required by Specification 4.0.5

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	61
2	43
3	26

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LOOP A	LOOP B	LOOP C	LOOP D	LIFT SETTING[1] (±3%)[2]
1.	PSV-7410	PVS-7420	PSV-7430	PSV-7440	1285 psig
2.	PSV-7410A	PVS-7420A	PSV-7430A	PSV-7440A	1295 psig
3.	PSV-7410B	PVS-7420B	PSV-7430B	PSV-7440B	1305 psig
4.	PSV-7410C	PVS-7420C	PSV-7430C	PSV-7440C	1315 psig
5.	PSV-7410D	PVS-7420D	PSV-7430D	PSV-7440D	1325 psig

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

[2]The as left lift setting shall be within ±1% following valve testing.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Three motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one motor-driven auxiliary feedwater pump inoperable, within 28 days restore the pump to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the turbine-driven auxiliary feedwater pump inoperable, or with any two auxiliary feedwater pumps inoperable, within 72 hours restore the affected auxiliary feedwater pump(s) to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. MODE 3 may be entered with an inoperable turbine-driven auxiliary feedwater pump for the purposes of performing post-maintenance testing and Surveillance Requirement 4.7.1.2.1.a.2.

c. With three auxiliary feedwater pumps inoperable, within 1 hour apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With four auxiliary feedwater pumps inoperable, immediately initiate action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible. LCO 3.0.3 and all other LCO actions requiring Mode changes are suspended until one of the four inoperable auxiliary feedwater pumps is restored to OPERABLE status.

e. Specification 3.0.4.b is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying that the developed head of each motor-driven pump at the flow test point is greater than or equal to the required developed head;

2) Verifying that the developed head of the steam turbine-driven pump at the flow test point is greater than or equal to the required developed head when tested at a secondary steam supply pressure greater than 1000 psig within 72 hours after entry into MODE 3;

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that each automatic valve in the flow path is in the correct position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER.

b. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal.

3) Verifying that each auxiliary feedwater flow regulating valve limits the flow to each steam generator between 550 gpm and 675 gpm.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater that 30 days prior to entering MODE 2 by verifying normal flow to each steam generator.

AUXILIARY FEEDWATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The auxiliary feedwater storage tank (AFST) shall be OPERABLE with a contained water volume of at least 485,000 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the AFST inoperable, within 4 hours restore the AFST to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The AFST shall be demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by verifying the contained water volume is within its limits.

PLANT SYSTEMS

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microCurie/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	At a frequency in accordance with the Surveillance Frequency Control Program
2. Isotopic Analysis for DOSE EQUIVALENT I −131 Concentration	a) At a frequency in accordance with the Surveillance Frequency Control Program, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) At a frequency in accordance with the Surveillance Frequency Control Program, whenever the gross radioacitivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

a. With one MSIV inoperable but open, POWER OPERATION may continue provided that within 4 hours the inoperable valve is restored to OPERABLE status or the requirements of the CRMP are met; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With more than one MSIV inoperable but open, POWER OPERATION may continue provided that within 1 hour at least three inoperable valves are restored to OPERABLE status or the requirements of the CRMP are met; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

PLANT SYSTEMS

ATMOSPHERIC STEAM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 At least four atmospheric steam relief valves shall be OPERABLE.

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

ACTION:

a. With one less than the required atmospheric steam relief valves OPERABLE, within 7 days restore the required atmospheric steam relief valves to OPERABLE status or apply the requirements of the CRMP; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours and place the required RCS/RHR loops in operation for decay heat removal.

b. With two less than the required atmospheric relief valves OPERABLE, within 72 hours restore at least three atmospheric relief valves to OPERABLE status or apply the requirements of the CRMP or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours and place the required RCS/RHR loops in operation for decay heat removal.

c. With more than two less than the required atmospheric relief valves OPERABLE, within 1 hour restore at least two atmospheric relief valves to OPERABLE status or apply the requirements of the CRMP or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours and place the required RCS/RHR loops in operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric relief valve shall be demonstrated OPERABLE prior to startup following any COLD SHUTDOWN of 30 days or longer or following any refueling shutdown, by verifying that all valves will open and close fully by operation of automatic# and manual controls.

* When steam generators are being used for decay heat removal.

Required to be met for automatic controls only in MODES 1 and 2.

PLANT SYSTEMS

MAIN FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.7 Each main feedwater isolation valve (MFIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODES 1 and 2: With one MFIV inoperable but open, operation may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours.

NOTE:
On a one-time basis, with the Unit 2 Train D MFIV inoperable but open, operation may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours. This note expires 30 days after approval of the license amendment that approved this change.

MODE 3: With one MFIV inoperable, subsequent operation in MODE 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.7 Each MFIV shall be demonstrated OPERABLE by verifying full closure within 10 seconds when tested pursuant to Specification 4.0.5. The provisions of specification 4.0.4 are not applicable for entry into MODE 3.

3.7.2 (This specification not used)

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least three independent component cooling water loops shall be OPERABLE.

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

ACTION:

a. With only two component cooling water loops OPERABLE, within 7 days restore at least three loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more component cooling water loops inoperable, within 1 hour restore at least two loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least three component cooling water loops shall be demonstrated OPERABLE:

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

b. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that:

   1) Each automatic valve servicing safety-related equipment or isolating the non-nuclear safety portion of the system actuates to its correct position on a Safety Injection, Loss of Offsite Power, Containment Phase "B" Isolation, or Low Surge Tank test signal, as applicable (performed during shutdown);

   2) Each Component Cooling Water System pump starts automatically on a Safety Injection or Loss of Offsite Power test signal (performed during shutdown); and

   3) The surge tank level instrumentation which provides automatic isolation of portions of the system is demonstrated OPERABLE by performance of a CHANNEL CALIBRATION test.

c. By verifying that each valve inside containment (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position prior to entering MODE 4 following each COLD SHUTDOWN of greater than 72 hours if not performed within the previous 31 days.

3/4.7.4 ESSENTIAL COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least three independent essential cooling water loops shall be OPERABLE.

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

ACTION:

a. With only two essential cooling water loops OPERABLE, within 7 days restore at least three loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more essential cooling water loops inoperable, within 1 hour restore at least two loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least three essential cooling water loops shall be demonstrated OPERABLE:

a. At a frequency 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. At a frequency in accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:

   1) Each Essential Cooling Water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal, and

   2) Each Essential Cooling Water pump starts automatically on an actual or simulated signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 25.5 feet Mean Sea Level, USGS datum, and

b. An Essential Cooling Water intake temperature of less than or equal to 99°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION is applicable to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by verifying the intake water temperature and water level to be within their limits.

3/4.7.6 (This specification number is not used.)

3/4.7.7 CONTROL ROOM MAKEUP AND CLEANUP FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Three independent Control Room Makeup and Cleanup Filtration Systems shall be OPERABLE.

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

ACTION:

a. With one Control Room Makeup and Cleanup Filtration System inoperable for reasons other than condition b or condition e, within 7 days restore the inoperable system to OPERABLE status, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one Control Room Makeup and Cleanup Filtration System inoperable only due to unavailability of cooling, within 7 days restore the inoperable system to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two Control Room Makeup and Cleanup Filtration Systems inoperable for reasons other than condition e, within 72 hours restore at least two systems to OPERABLE status, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With three Control Room Makeup and Cleanup Filtration Systems inoperable for reasons other than condition e, within 12 hours restore at least one system to OPERABLE status, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Note

The Control Room Envelope (CRE) boundary is not a required system, subsystem, train, component, or device that depends on a diesel generator as a source of emergency power. Specification 3.8.1.1.d need not be applied for an inoperable Control Room Makeup and Cleanup Filtration System that is inoperable solely due to an inoperable Control Room Envelope boundary.

e. One or more Control Room Makeup and Cleanup Filtration Systems inoperable due to inoperable Control Room Envelope (CRE) boundary perform the following:

1) immediately initiate action to implement mitigating actions, and
2) within 24 hours verify mitigating actions ensure CRE occupant exposures to radiological, chemical and smoke hazards will not exceed limits, and
3) within 90 days restore CRE boundary to OPERABLE status.

OR

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

SURVEILLANCE REQUIREMENTS

4.7.7 Each Control Room Makeup and Cleanup Filtration System shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 78°F;

b. At a frequency in accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers of the makeup and cleanup air filter units and verifying that the system operates for at least 15 continuous minutes with the makeup filter unit heaters operating;

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

   1) Verifying that the makeup and cleanup systems satisfy the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% for HEPA filter banks and 0.10% for charcoal adsorber banks and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%; and

   3) Verifying a system flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 for a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and a relative humidity of 70%.

e. At a frequency in accordance with the Surveillance Frequency Control Program by:

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.1 inches Water Gauge for the makeup units and 6.0 inches Water Gauge for the cleanup units while operating the system at a flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units;

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that on a control room emergency ventilation test signal (High Radiation and/or Safety Injection test signal), the system automatically switches into a recirculation and makeup air filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks of the cleanup and makeup units;

3) Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program; and

4) Verifying that the makeup filter unit heaters dissipate 4.5 ± 0.45 kW when tested in accordance with ANSI N510-1980.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.10% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 6000 cfm ± 10% for the cleanup units and 1000 cfm ± 10% for the makeup units.

3/4.7.8 through 3/4.7.13 (These specification numbers are not used)

Pages 3/4 7-20 through 3/4 7-32 have been deleted

3/4.7.14 ESSENTIAL CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.14 At least three independent Essential Chilled Water System loops shall be OPERABLE.

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

ACTION:

a. With only two Essential Chilled Water System loops OPERABLE, within 7 days restore at least three loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more Essential Chilled Water System loops inoperable, within 1 hour restore at least two loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.14 The Essential Chilled Water System shall be demonstrated OPERABLE by:

a. Performance of surveillances as required by Specification 4.0.5, and

b. At a frequency in accordance with the Surveillance Frequency Control Program by demonstrating that the system starts automatically on a Safety Injection test signal.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System (1), and

b. Three separate and independent standby diesel generators, each with a separate fuel tank containing a minimum volume of 60,500 gallons of fuel, and an automatic load sequencer.

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

ACTION:

a. With one offsite circuit of the above-required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Within 72 hours restore the offsite circuit to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a standby diesel generator inoperable, demonstrate the OPERABILITY of the above-required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the standby diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE standby diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.5) for each such standby diesel generator separately within 8 hours, unless it can be demonstrated there is no common mode failure for the remaining diesel generator(s). Within 14 days restore the inoperable standby diesel generator to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. (12)

c. With one offsite circuit of the above-required A.C. electrical power sources and one standby diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.1.a. within 1 hour and at least once per 8 hours thereafter; and if the standby diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive

ACTION (Continued)

maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE standby diesel generator(s) by performing Surveillance Requirement 4.8.1.1.2.a.5) within 8 hours, unless it can be demonstrated there is no common mode failure for the remaining diesel generators; within 12 hours restore at least one of the inoperable sources to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (12)

d. With one standby diesel generator inoperable in addition to ACTION b. or c. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generators as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

   If these conditions are not satisfied within 24 hours, apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and In COLD SHUTDOWN within the following 30 hours.

e. With two of the above required offsite A.C. circuits inoperable, within 24 hours restore at least one of the inoperable offsite sources to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours.

f. With two or three of the above required standby diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1.a. within 1 hour and at least once per 8 hours thereafter. With three of the above required standby diesel generators inoperable, within 2 hours restore at least one standby diesel generator to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (12) With two of the above required standby diesel generators inoperable, within 24 hours restore at least two standby diesel generators to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION

ACTION (continued)

g. With one or more diesel generator fuel oil storage tanks with stored fuel oil total particulates not within the Diesel Fuel Oil Testing Program limits, within 7 days restore the fuel oil total particulates within limits, or declare the associated standby diesel generator(s) inoperable.

h. With one or more diesel generator fuel oil storage tanks with new fuel oil properties not within the Diesel Fuel Oil Testing Program limits, within 30 days restore the fuel oil properties within limits, or declare the associated standby diesel generator(s) inoperable.

i. Specification 3.0.4.b is not applicable for standby diesel generators.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program during shutdown by transferring the unit power supply from the normal circuit to each of the alternate circuits.

4.8.1.1.2 Each standby diesel generator shall be demonstrated OPERABLE: (2)(11)

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying the fuel level in its associated fuel tank,

2) Verifying the diesel starts from standby condition and achieves a voltage and frequency of 4160 ± 416 volts and 60 ± 1.2 Hz (3). The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-off site power by itself, or

c) Simulated loss-of-offsite power in conjunction with a Safety Injection test signal, or

d) A Safety Injection test signal by itself.

3) Verifying the generator is synchronized, loaded to 5000 to 5500 kW, and operates with a load of 5000 to 5500 kW for at least 60 minutes, (4)(6) and

4) Verifying the standby diesel generator is aligned to provide standby power to the associated emergency busses.

5) Verifying the diesel starts from standby conditions and accelerates to 600 rpm (nominal) in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The steady-state voltage and frequency shall be 4160 ± 208 volts and 60 ± 0.3 Hz. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-off site power by itself, or

c) Simulated loss-of-offsite power in conjunction with a Safety Injection test signal, or

d) A Safety Injection test signal by itself.

b. At a frequency in accordance with the Surveillance Frequency Control Program and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from its associated fuel tank;

c. Maintain properties of new and stored fuel oil in accordance with the Fuel Oil Monitoring Program.

SURVEILLANCE REQUIREMENTS (Continued)

d. Deleted

e. At a frequency in accordance with the Surveillance Frequency Control Program, during shutdown, by:

1) Deleted

2) Verifying the generator capability to reject a load of greater than or equal to 785.3 kW while maintaining voltage at 4160 ± 416 volts and frequency at 60 ± 4.5 Hz; (4)(5)

3) Verifying the generator capability to reject a load of 5500 kW without tripping. The generator voltage shall not exceed 5262 volts during and following the load rejection; (4)(5)

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the ESF busses and load shedding from the ESF busses, and

b) Verifying the diesel starts on the auto-start signal within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 ± 208 volts and 60 ± 0.3 Hz during this test.

5) Verifying that on a Safety Injection test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the autostart signal; the steady-state generator voltage and frequency shall be maintained at 4160 ± 208 volts and 60 ± 0.3 Hz;

6) Simulating a loss-of-offsite power in conjunction with a Safety Injection test signal, and:

a) Verifying deenergization of the ESF busses and load shedding from the ESF busses;

b) Verifying the diesel starts on the auto-start signal within 10 seconds, energizes the auto-connected ESF (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the ESF loads. After energization, the steady-state voltage and frequency of the ESF busses shall be maintained at 4160 ± 208 volts and 60 ± 0.3 Hz during this test; and

c) Verifying that all automatic diesel generator trips, except engine overspeed, generator differential, and low lube oil pressure are automatically bypassed upon loss of voltage on the ESF bus concurrent with a Safety Injection Actuation signal.

7)(10) Verifying the standby diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to 5700 to 6050 kW (4)(5)(6) and during the remaining 22 hours of this test, the diesel generator shall be loaded to 5000 to 5500 kW. (6) The steady-state generator voltage and frequency shall be 4160 ± 208 volts and 60 ± 0.3 Hz during this test. Within 5 minutes after completing this 24-hour test, perform a fast start per Specification 4.8.1.1.2.a.5) (7);

8) Verifying that the auto-connected loads to each standby diesel generator do not exceed the 2000-hour rating of 5935 kW;

9) Verifying the standby diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its ESF loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

10) Verifying that with the standby diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the ESF loads with offsite power; (5)

11) Verifying that the automatic load sequence timer is OPERABLE with the first sequenced load verified to be loaded between 1.0 second and 1.6 seconds, and all other load blocks within ±10% of its design interval;

12) Verifying that the standby diesel generator emergency stop lockout feature prevents diesel generator starting; and

13) Demonstrating the OPERABILITY of the automatic load shed bypass and the manual load shed reinstatement features of the load sequencer.

f. At a frequency in accordance with the Surveillance Frequency Control Program or after any modifications which could affect standby diesel generator interdependence by starting all standby diesel generators simultaneously, during shutdown, and verifying that all standby diesel generators accelerate to at least 600 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The steady-state voltage and frequency shall be maintained at 4160 ± 208 volts and 60 ± 0.3 Hz.

g. At a frequency in accordance with the Surveillance Frequency Control Program by draining each fuel tank, removing the accumulated sediment and cleaning the tank.

4.8.1.1.3 (Not used)

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

(Not used)

SPECIFICATION NOTATIONS

(1) Loss of one 13.8 kV Standby Bus to 4.16 kV ESF bus line constitutes loss of one offsite source. Loss of two 13.8 kV Standby busses to 4.16 kV ESF bus lines constitutes loss of two offsite sources.

(2) All diesel generator starts for the purpose of these surveillances may be preceded by a prelube period.

(3) The diesel generator start for this surveillance may be a modified start involving reduced fuel (load limit) and/or idling and gradual acceleration to synchronous speed.

(4) Generator loading may be accomplished in accordance with vendor recommendations, including a warmup period prior to loading.

(5) The diesel generator start for this surveillance may be a modified start (see SR 4.8.1.1.2.a.2)).

(6) Momentary transients outside this load range due to changing conditions on the grid shall not invalidate the test.

(7) If Specification 4.8.1.1.2.a.5) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the standby diesel generator may be operated at 5000-5500 kW for a minimum of 2 hours or until operating temperature has stabilized.

(8) (Not used)

(9) (Not used)

(10) This test may be performed during power operation provided that the other two diesel generators are operable.

(11) Credit may be taken for events that satisfy any of these Surveillance Requirements.

(12) For the Unit 2 Train B standby diesel generator (SDG 22) failure of December 9, 2003, restore the inoperable standby diesel generator to OPERABLE status within 113 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

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ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. Two[1] standby diesel generators each with a separate fuel tank containing a minimum volume of 60,500 gallons of fuel.

APPLICABILITY: MODE 5 and MODE 6 with water level in the refueling cavity <23 ft above the reactor pressure vessel flange.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, movement of irradiated fuel, operations with a potential for draining the reactor vessel or crane operation with loads over the spent fuel pool. Immediately initiate actions to restore the inoperable A.C. electrical power source to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.3), and 4.8.1.1.3.

4.8.1.2.1 The alternate onsite emergency power source shall be demonstrated functional by:

a. Within 4 hours of taking credit for the onsite emergency power source as a standby diesel generator, verify it starts and achieves steady state voltage (±10%) and frequency (±2%) in 5 minutes.

b. Within 4 hours of taking credit for the onsite emergency power source as a standby diesel generator and every 8 hours thereafter, verify the emergency power source is capable of being aligned to the required ESF bus by performing a breaker alignment check.

[1]An alternate onsite emergency power source, capable of supplying power for one train of shutdown cooling may be substituted for one of the required diesels for 14 consecutive days (SR 4.8.1.2.1 is the only requirement applicable).

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.3 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 standby diesel generator with a separate fuel tank containing a minimum volume of 60,500 gallons of fuel.

APPLICABILITY: MODE 6 with water level in the refueling cavity $\geq$ 23 ft above the reactor pressure vessel flange.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, or operations with a potential for draining the reactor vessel. Immediately initiate actions to restore the inoperable A.C. electrical power source to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.3 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.3), and 4.8.1.1.3.

ELECTRICAL POWER SYSTEMS

3/4.8.2 DC SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following DC electrical sources shall be OPERABLE:

a. Channel I 125-volt Battery Bank E1A11 (Unit 1), E2A11 (Unit 2) and one of its two associated chargers,

b. Channel II 125-volt Battery Bank E1D11 (Unit 1), E2D11 (Unit 2) and one of its two associated full capacity chargers,

c. Channel III 125-volt Battery Bank E1B11 (Unit 1), E2B11 (Unit 2) and one of its two associated full capacity chargers, and

d. Channel IV 125-volt Battery Bank E1C11 (Unit 1), E2C11 (Unit 2) and one of its two associated chargers.

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

ACTION:

NOTE
If the batteries discharge for more than 2 hours as the sole source of power to their DC bus while the CRMP is being applied and no alternate source of power is available, the LCO shall be considered not met.

a. With one of the required battery banks inoperable, within 2 hours restore the inoperable battery bank to OPERABLE status or apply the requirements of the CRMP or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With more than one of the required battery banks inoperable, within 1 hour restore at least three battery banks to OPERABLE status or apply the requirements of the CRMP or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one channel with no battery chargers OPERABLE,

1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, AND

2. Verify float current for the affected battery does not exceed 2 amps once per 12 hours, AND

3. Restore one battery charger to OPERABLE status within 72 hours.

If the battery terminal voltage cannot be restored in the allowed time, float current is excessive, or a battery charger is not restored to operability in the time allowed, apply the requirements of the CRMP or the affected reactor unit is to be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS (Continued)

d. With more than one channel with no battery chargers OPERABLE,

1 Restore terminal voltage for at least three battery banks to greater than or equal to the minimum established float voltage within 1 hour, AND

2. Verify float current for the affected batteries does not exceed 2 amps once per 12 hours, AND

3. Restore one battery charger to OPERABLE status on at least three channels within 1 hour.

If the battery terminal voltage cannot be restored in the allowed time, float current is excessive, or a battery charger is not restored to operability in the time allowed, apply the requirements of the CRMP or the affected reactor unit is to be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With one of the required channels inoperable for reasons other than (a), (b), (c), or (d) above, restore the channel to OPERABLE status within 2 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that:

The total battery terminal voltage is greater than or equal to the minimum established float voltage.

b. Not used.

c. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that:

1. The battery charger can supply at least 300 amperes at greater than or equal to the minimum established float voltage for at least 8 hours.

OR

Each battery charger can recharge the battery to the fully charged state within 12 hours while supplying the largest combined demands of the various continuous steady-state loads following a battery discharge to the bounding design-basis event discharge state.

2.

NOTE:	
1.	The modified performance discharge test in SR 4.8.2.3.f may be performed in lieu of Surveillance Requirement 4.8.2.1.c.2.
2.	Credit may be taken for unplanned events that satisfy this surveillance requirement.

The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated ESF loads for the design duty cycle when the battery is subjected to a battery service test.

d. Not used.

e. Not used.

f. Not used.

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ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.3.2, "Onsite Power Distribution - Shutdown."

APPLICABILITY: MODES 5 and 6

ACTION:

a. With one or more required DC electrical power subsystems inoperable:

1. Immediately declare affected required feature(s) inoperable, OR

2. Immediately:

- Initiate action to suspend operation with a potential for draining the reactor vessel, AND

- Suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, AND

- Initiate corrective action to restore the required DC electrical power subsystems to OPERABLE status as soon as possible.

b. With no battery chargers for a required channel OPERABLE:

1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, AND

2. Verify float current for the affected battery does not exceed 2 amps once per 12 hours, AND

3. Restore one battery charger to OPERABLE status within 72 hours.

If the battery terminal voltage cannot be restored within the allowed time, float current is excessive, or a battery charger is not restored to operability in the time allowed:

- Initiate action to suspend operation with a potential for draining the reactor vessel, AND

LIMITING CONDITION FOR OPERATION (continued)

- Suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, AND
- Initiate corrective action to restore the required DC electrical power subsystems to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 Each 125-volt battery bank shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the total battery terminal voltage is greater than or equal to the minimum established float voltage.

b. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the battery charger can supply at least 300 amperes at greater than or equal to the minimum established float voltage for at least 8 hours.

OR:

Verify each battery charger can recharge the battery to the fully charged state within 12 hours while supplying the largest combined demands of the various continuous steady-state loads following a battery discharge to the bounding design-basis event discharge state.

c.

NOTE:	1.	The modified performance discharge test in SR 4.8.2.3.f may be performed in lieu of Surveillance Requirement 4.8.2.2.c.
	2.	Credit may be taken for unplanned events that satisfy this surveillance requirement.

At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated ESF loads for the design duty cycle when the battery is subjected to a battery service test.

ELECTRICAL POWER SYSTEMS

BATTERY PARAMETERS

LIMITING CONDITION FOR OPERATION

3.8.2.3 Parameters for the Class 1E batteries shall be within the specified limits.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTION:

a. If a battery has one or more cells with float voltage < 2.07 V:

1. Perform surveillance requirement 4.8.2.1.a within 2 hours, AND
2. Perform surveillance requirement 4.8.2.3.a within 2 hours, AND
3. Restore float voltage of the affected cell(s) to ≥ 2.07 volts within 24 hours.

If the required action and associated completion of the above conditions are not met, declare the associated battery INOPERABLE immediately.

b. If a battery has float current > 2 amps:

1. Perform surveillance requirement 4.8.2.1.a within 2 hours, AND
2. Restore battery float current to ≤ 2 amps within 12 hours.

If the required action and associated completion of the above conditions are not met, declare the associated battery INOPERABLE immediately.

c. If a battery has one or more cells with electrolyte level less than minimum established design limits:

1. Restore electrolyte level in the affected cell(s) to above the top of the plates within 8 hours if electrolyte level is below the top of the plates, AND
2. Verify there is no evidence of electrolyte leakage within 12 hours if electrolyte level is below the top of the plates, AND
3. Restore electrolyte level in the affected cell(s) to greater than or equal to minimum established design limits within 31 days.

If the required action and associated completion of the above conditions are not met, declare the associated battery INOPERABLE immediately.

d. If a battery has a 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.

If the required action and associated completion of the above conditions are not met, declare the associated battery INOPERABLE immediately.

e. If battery parameters are not within limits for 2 or more batteries, restore battery parameters to within design limits within 2 hours with no more than one battery outside design limits if a longer time for completion is applicable.

If the required action and associated completion of the above conditions are not met, declare the associated batter(ies) INOPERABLE immediately.

LIMITING CONDITION FOR OPERATION (continued)

f. If a battery has one or more battery cells with float voltage < 2.07 volts and float current > 2 amps, declare the associated battery INOPERABLE immediately.

SURVEILLANCE REQUIREMENTS

4.8.2.3. Each 125-volt battery bank and charger shall be demonstrated operable:

a.

> NOTE: Performance of this surveillance is not required when battery terminal voltage is less than the minimum established float voltage of surveillance requirement 4.8.2.1.a.

At a frequency in accordance with the Surveillance Frequency Control Program, verify the float current for each battery is ≤ 2 amps.

b. At a frequency in accordance with the Surveillance Frequency Control Program, verify each battery pilot cell voltage is ≥ 2.07 V on float charge.

c. At a frequency in accordance with the Surveillance Frequency Control Program, verify each battery connected cell voltage is ≥ 2.07 V on float charge.

d. At a frequency in accordance with the Surveillance Frequency Control Program, verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.

e. At a frequency in accordance with the Surveillance Frequency Control Program, verify each battery pilot cell temperature is greater than or equal to minimum established design limits.

f. Battery capacity is tested under the following conditions:

1. At least once per 12 months by giving modified performance discharge tests of battery capacity to any battery that shows degradation or reaches 85% of the service life expected for the application with capacity less than 100% of the manufacturer's rating. Degradation is indicated when battery capacity drops more than 10% from its capacity on the previous performance/modified performance discharge test, or is below 90% of the manufacturer's rating; AND

2. At least once per 24 months by giving modified performance discharge tests of battery capacity to any battery reaching 85% of the service life with capacity greater than or equal to 100% of the manufacturer's rating; AND

3. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a modified performance discharge test.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner:

a. Train A A.C. ESF Busses consisting of:
   1) 4160-Volt ESF Bus # E1A (Unit 1), E2A (Unit 2), and
   2) 480-Volt ESF Busses # E1A1 and E1A2 (Unit 1), E2A1 and E2A2 (Unit 2) from respective load center transformers.

b. Train B A.C. ESF Busses consisting of:
   1) 4160-Volt ESF Bus # E1B (Unit 1), E2B (Unit 2), and
   2) 480-Volt ESF Busses # E1B1 and E1B2 (Unit 1), E2B1 and E2B2 (Unit 2) from respective load center transformers.

c. Train C A.C. ESF Busses consisting of:
   1) 4160-Volt ESF Bus # E1C (Unit 1), E2C (Unit 2), and
   2) 480-Volt ESF Busses # E1C1 and E1C2 (Unit 1), E2C1 and E2C2 (Unit 2) from respective load center transformers.

d. 120-Volt A.C. Vital Distribution Panels DP1201 and DP001 energized from their associated inverters connected to D.C. Bus # E1A11* (Unit 1), E2A11* (Unit 2),

e. 120-Volt A.C. Vital Distribution Panel DP1202 energized from its associated inverter connected to D.C. Bus # E1D11* (Unit 1), E2D11* (Unit 2),

f. 120-Volt A.C. Vital Distribution Panel DP1203 energized from its associated inverter connected to D.C. Bus # E1B11* (Unit 1), E2B11* (Unit 2),

g. 120-Volt A.C. Vital Distribution Panels DP1204 and DP002 energized from their associated inverters connected to D.C. Bus # E1C11* (Unit 1), E2C11* (Unit 2),

h. 125-Volt D.C. Bus E1A11 (Unit 1) E2A11 (Unit 2) energized from Battery Bank E1A11 (Unit 1), E2A11 (Unit 2),

i. 125-Volt D.C. Bus E1D11 (Unit 1) E2D11 (Unit 2) energized from Battery Bank E1D11 (Unit 1), E2D11 (Unit 2),

j. 125-Volt D.C. Bus E1B11 (Unit 1) E2B11 (Unit 2) energized from Battery Bank E1B11 (Unit 1), E2B11 (Unit 2), and

k. 125-Volt D.C. Bus E1C11 (Unit 1) E2C11 (Unit 2) energized from Battery Bank E1C11 (Unit 1), E2C11 (Unit 2).

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*The inverter(s) associated with one channel may be disconnected from its D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on its associated battery bank provided: (1) its vital distribution panels are energized, and (2) the vital distribution panels associated with the other battery banks are energized from their associated inverters and connected to their associated D.C. busses.

LIMITING CONDITION FOR OPERATION (Continued)

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

ACTION:

a. With one of the required trains of A.C. ESF busses not fully energized, within 8 hours reenergize the train or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With more than one of the required trains of A.C. ESF busses not fully energized, within 1 hour reenergize at least two trains or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. vital distribution panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) within 2 hours reenergize the A.C. distribution panel or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) within 24 hours reenergize the A.C. vital distribution panels from its associated inverter connected to its associated D.C. bus or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With more than one A.C. vital distribution panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) within 1 hour reenergize at least five A.C. distribution panels or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) within 1 hour reenergize at least five A.C. vital distribution panels from their associated inverter connected to their associated D.C. bus or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With one D.C. bus not energized from its associated battery bank, within 2 hours reenergize the D.C. bus from its associated battery bank or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With more than one D.C. bus not energized from its associated battery bank, within 1 hour reenergize at least three D.C. buses from their associated battery banks or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at a frequency in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 The necessary portion of AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With one or more required AC, DC, or AC vital bus electrical power distribution subsystems inoperable, immediately declare associated supported required feature(s) inoperable OR immediately initiate action to suspend operations with a potential for draining the reactor vessel, suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration, and immediately initiate corrective action to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status and declare associated required residual heat removal subsystem(s) inoperable and not in operation.

SURVEILLANCE REQUIREMENT

4.8.3.2 Verify correct breaker alignment and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems at a frequency in accordance with the Surveillance Frequency Control Program.

3.8.4 (This specification number is not used.)

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

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

b. A boron concentration of greater than or equal to 2800 ppm, and

c. Each valve or mechanical joint used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.*

ACTION:

a. With the requirements of LCO a. or b. not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2800 ppm, whichever is the more restrictive.

b. With a valve or mechanical joint used to isolate an unborated water source not secured in the closed position, immediately suspend CORE ALTERATIONS and initiate action to secure the valve(s) or mechanical joint(s) in the closed position and within 4 hours verify boron concentration is within limit. The required action to verify the boron concentration within limits must be completed whenever ACTION b. is entered. A separate ACTION entry is allowed for each unsecured valve or mechanical joint.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at a frequency in accordance with the Surveillance Frequency Control Program.

4.9.1.3 Each valve or mechanical joint used to isolate unborated water sources shall be verified closed and secured in position at a frequency in accordance with the Surveillance Frequency Control Program.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors* shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at a frequency in accordance with the Surveillance Frequency Control Program,

b. A CHANNEL CALIBRATION, excluding the Neutron detectors, at a frequency in accordance with the Surveillance Frequency Control Program.

* An Extended Range Neutron Flux Monitor may be substituted for one of the Source Range Neutron Flux Monitors provided the OPERABLE Source Range Neutron Flux Monitor is capable of providing audible indication in the containment and control room.

3/4.9.3 through 3/4.9.7 (These specification numbers are not used)

Pages 3/4 9-4 through 3/4 9-7 have been deleted

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit of LCO 3.9.1 and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at a frequency in accordance with the Surveillance Frequency Control Program.

* The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit of LCO 3.9.1 and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at a frequency in accordance with the Surveillance Frequency Control Program.

*Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.9 (This specification is not used)

3/4.9.10 WATER LEVEL - REFUELING CAVITY

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods* within the refueling cavity when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at a frequency in accordance with the Surveillance Frequency Control Program thereafter during movement of fuel assemblies or control rods.

* Water level requirements are not applicable when control rods are moved in conjunction with the head package during a rapid refueling.

3/4.9.11 WATER LEVEL - STORAGE POOLS

SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.11.1 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:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILIANCE REQUIREMENTS

4.9.11.1 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at a frequency in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the spent fuel pool.

IN-CONTAINMENT STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11.2 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 in-containment storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11.2 The water level in the in-containment storage pool shall be determined to be at least its minimum required depth at a frequency in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the in-containment storage pool.

3/4.9.12 (This specification is not used)

Pages 3/4 9-15 and 3/4 9-16 have been deleted

REFUELING OPERATIONS

3/4.9.13 SPENT FUEL POOL MINIMUM BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.13 The boron concentration of the spent fuel pool water shall be maintained greater than or equal to 2500 ppm.

APPLICABILITY: Whenever one or more fuel assemblies are stored in the spent fuel pool racks.

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of fuel assemblies in the spent fuel storage pool and initiate action to restore the boron concentration in the spent fuel pool to greater than or equal to 2500 ppm.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 The boron concentration of the spent fuel pool shall be determined by chemical analysis at a frequency in accordance with the Surveillance Frequency Control Program.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, initiate boration within 15 minutes and continue boration until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, initiate boration within 15 minutes and continue boration until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at a frequency in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at a frequency in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at a frequency in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

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

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

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

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at a frequency in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F at a frequency in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of STARTUP and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at a frequency in accordance with the Surveillance Frequency Control Program during STARTUP and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

3/4.10.5 (This specification not used)

3.10.6 (This specification not used)

3.10.7 (This specification not used)

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each unprotected outdoor tank shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any unprotected outdoor tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each unprotected outdoor tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at a frequency in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

*Tanks included in this specification are those outdoor tanks that are either not surrounded by liners, dikes, or walls capable of holding the tank contents or that do not have tank overflows and surrounding area drains connected to the Liquid Waste Processing System.

3/4.11.2 GASEOUS EFFLUENTS

3.11.2.5 (This specification not used)

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 1.0×10^5 Curies of noble gases (considered as Xe -133 equivalent)

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding above limit, immediately suspend all additions of the radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at a frequency in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

INTENTIONALLY BLANK

SECTION 3/4.12 DELETED IN ITS ENTIRETY

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4.

The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not includes areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonable achievable, pursuant to 10 CFR 50.36a.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel-lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 150 feet.

b. Nominal inside height = 241.25 feet.

c. Minimum thickness of concrete walls = 4 feet.

d. Minimum thickness of concrete roof = 3 feet.

e. Minimum thickness of concrete floor mat = 18 feet.

f. Nominal thickness of steel liner = 3/8 inches.

g. Net free volume = 3.38×10^6 - 3.41×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 56.5 psig and a structural temperature of 286°F.

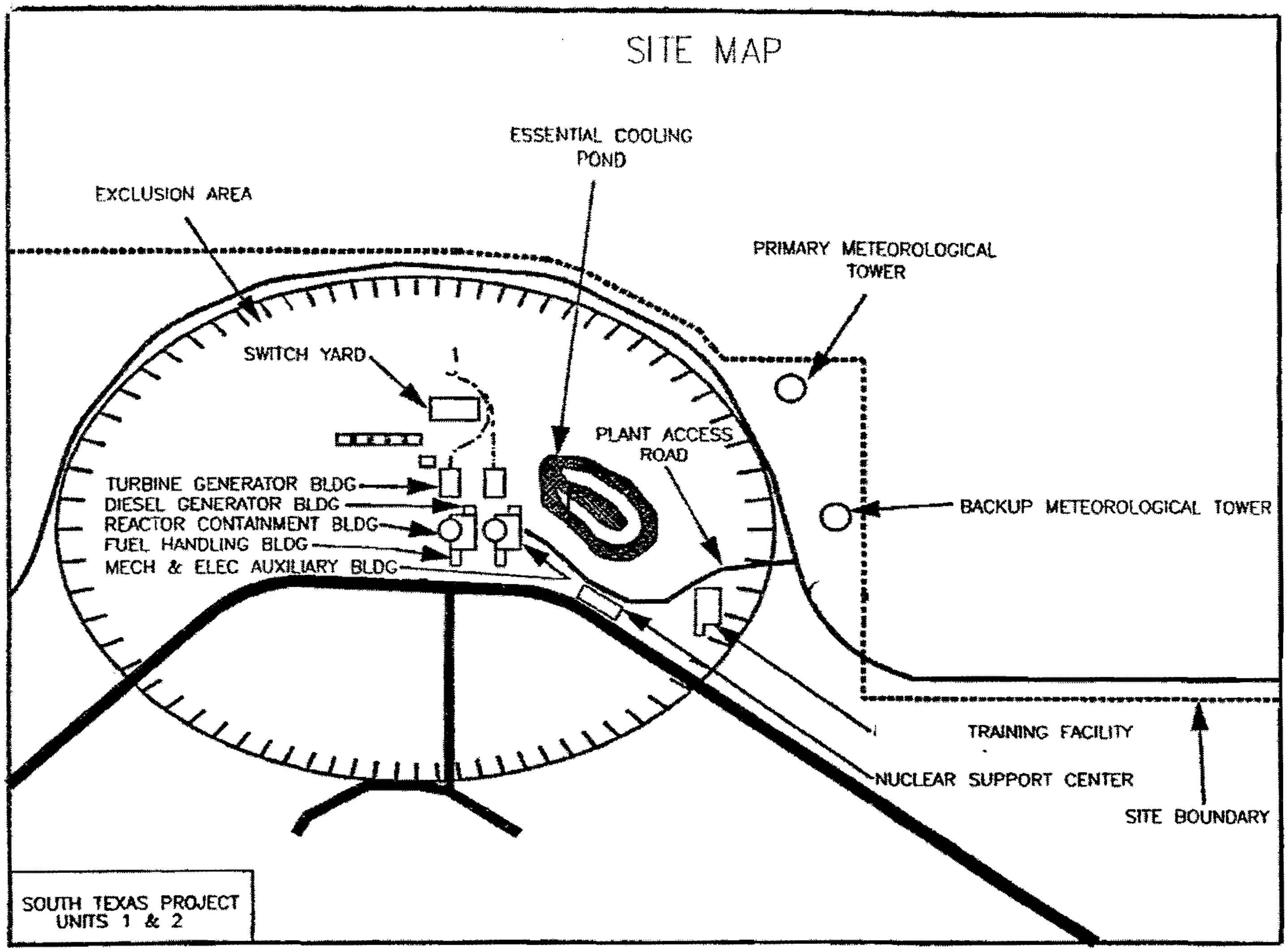


FIGURE 5.1-1
EXCLUSION AREA

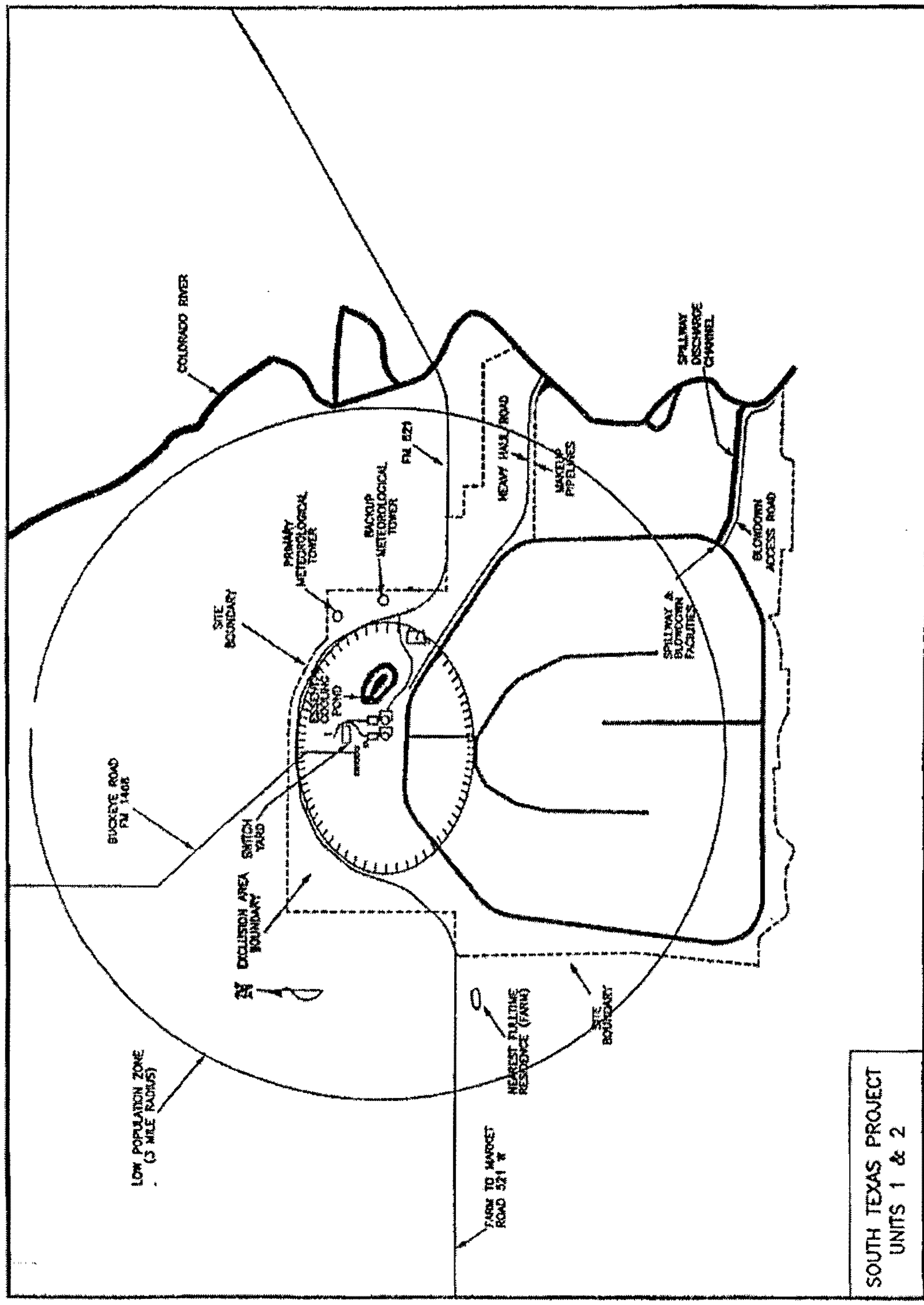


FIGURE 5.1-2
LOW POPULATION AREA

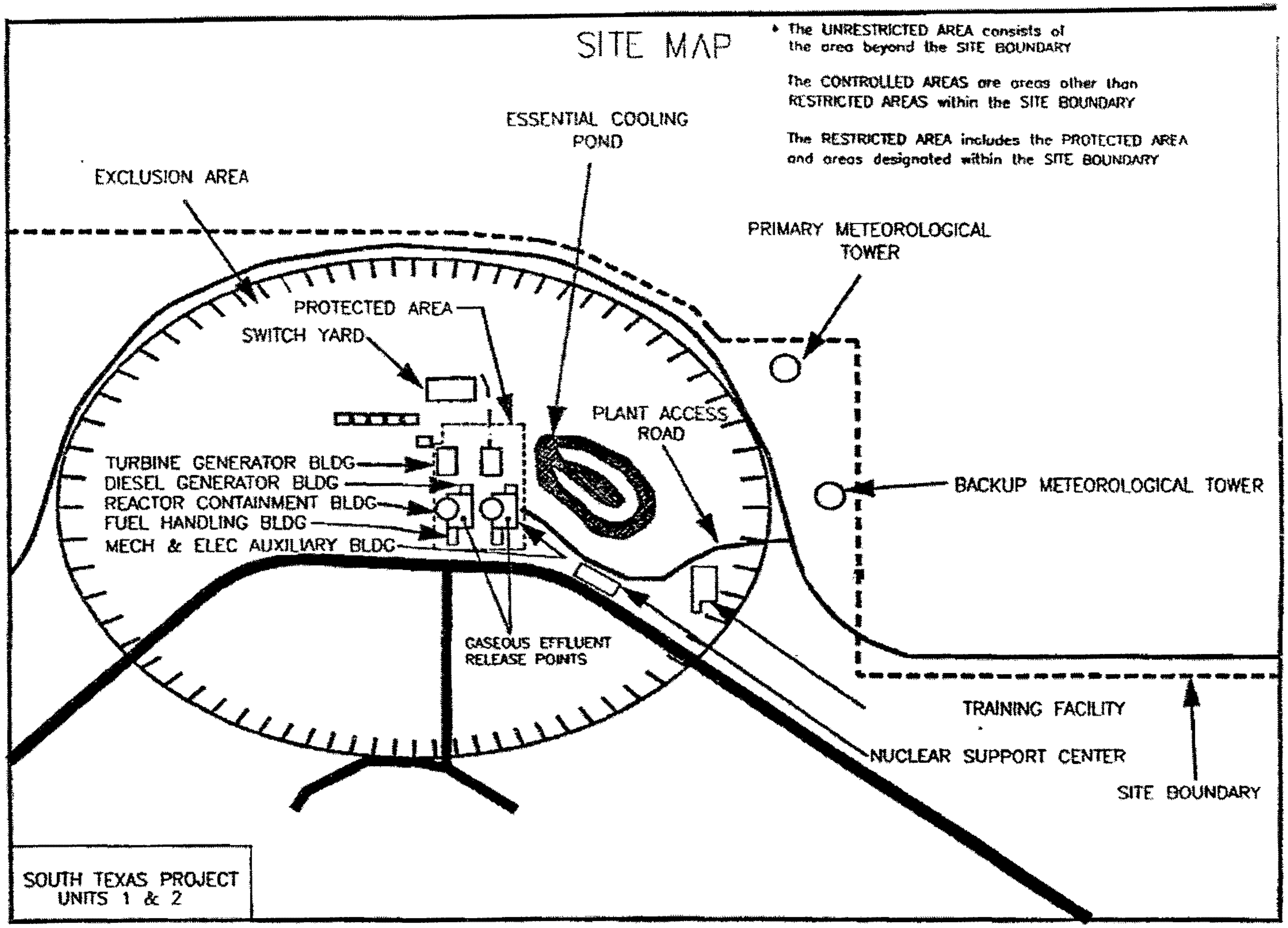


FIGURE 5.1-3
UNRESTRICTED AREA* AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS EFFLUENTS
(SEE ALSO FIGURE 5.1-4)

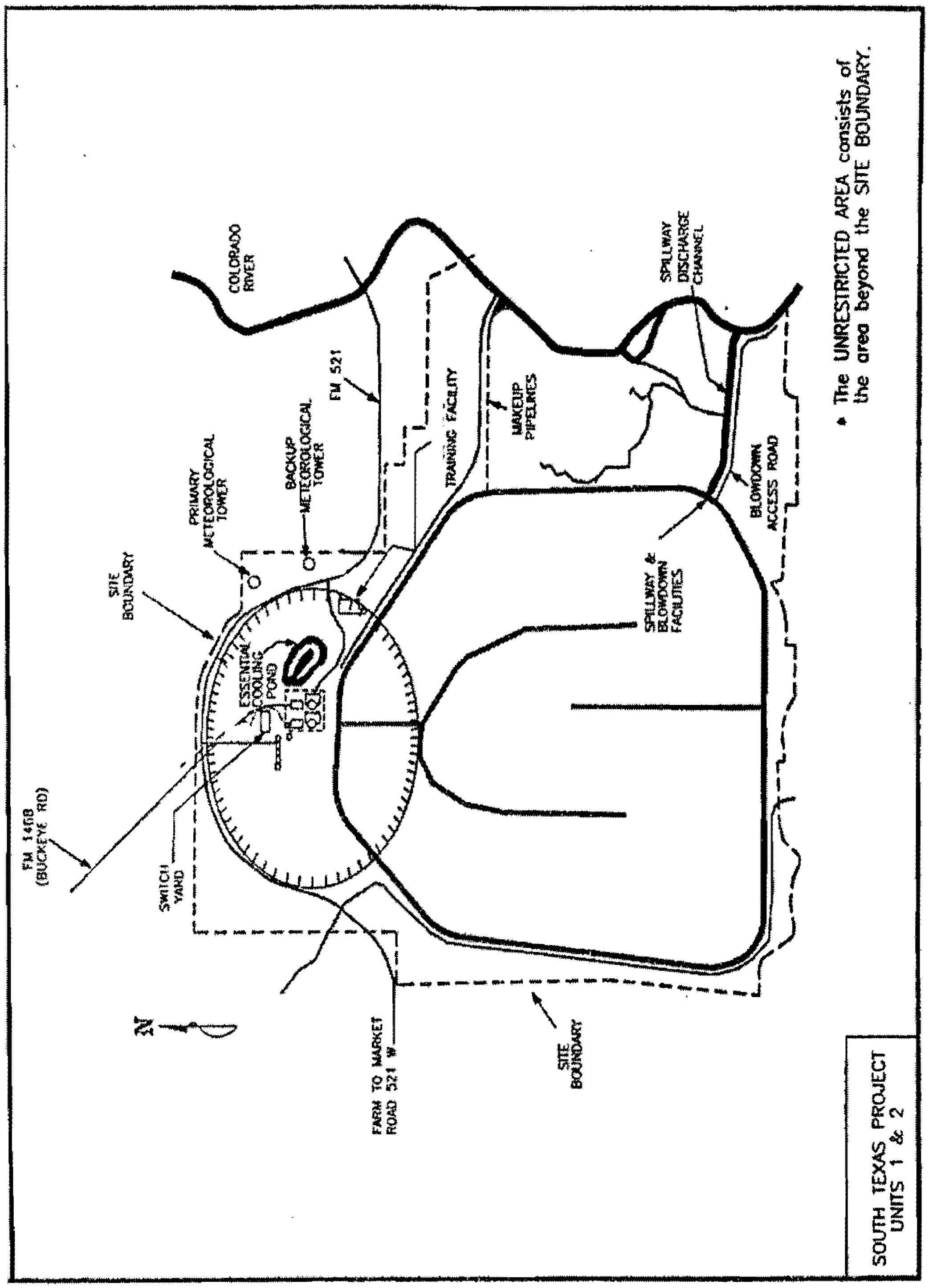


FIGURE 5.1-4

UNRESTRICTED AREA* AND SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies. Each fuel assembly shall consist of a matrix of zircaloy, ZIRLO™ or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy, ZIRLO™ or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The Unit 1 core shall contain 56 full-length control rod assemblies with no full-length control rod assembly installed in core location D-6. The Unit 2 core shall contain 57 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 158.9 inches of absorber material. The absorber material within each assembly shall be silver-indium-cadmium or hafnium. Mixtures of hafnium and silver-indium-cadmium are not permitted within a bank. All control rods shall be clad with stainless steel tubing.

5.4 (NOT USED)

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological towers shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. k_{eff} less than 1.0 when flooded with unborated water, which includes an allowance for uncertainties as described in WCAP-14416-NP-A.

b. k_{eff} less than or equal to 0.95 when flooded with water borated to 700 ppm, which includes an allowance for uncertainties as described in WCAP-14416-NP-A.

c. These requirements (a and b above) shall be met by storing fuel in the spent fuel storage racks according to Specifications 5.6.1.3, 5.6.1.4, and 5.6.1.5. Additionally, credit may be taken for the presence of soluble boron in the spent fuel pool water, per Specification 3.9.13, to mitigate the misloading of one or more fuel assemblies, as described in Specification 5.6.1.6.

d. A nominal 10.95 inches center to center distance between fuel assemblies in Region 1 of the storage racks and a nominal 9.15 inches center to center distance between fuel assemblies in Region 2 of the storage racks.

5.6.1.2 Prior to insertion into the spent fuel storage racks, each fuel assembly shall be categorized by reactivity, as discussed below, or be designated as a Category 1 fuel assembly. All fuel enrichment values are initial nominal uranium-235 enrichments. The reactivity categories are:

CATEGORY 1:
Fuel in Category 1 shall have an initial nominal enrichment of less than or equal to 4.95 w/o.

CATEGORY 2:
Fuel in Category 2 shall meet at least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 4.85 w/o; or,
2) a minimum burnup as shown on Figure 5.6-1.

CATEGORY 3:

Fuel in Category 3 shall meet at least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 3.55 w/o; or,
2) a minimum burnup as shown on Figure 5.6-2; or,
3) a minimum number of Integral Fuel Burnable Absorber (IFBA) pins, as shown on Figure 5.6-3.

The IFBA pin requirements shown in Figure 5.6-3 are based on nominal IFBA linear B^{10} loading of 1.57 mg-B^{10}/inch (1.0X). For higher IFBA loadings up to 2.35 mg-B^{10}/inch (1.5X), the required number of IFBA pins per assembly may be reduced by the ratio of the increased B^{10} loading to the nominal 1.57 mg-B^{10}/inch loading. A full length IFBA is 168 inches long, and a part length IFBA is greater than 120 inches long.

CATEGORY 4:

Fuel in Category 4 shall meet at least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 2.50 w/o; or,
2) an minimum burnup as shown in Figure 5.6-4; or,
3) a minimum number of IFBA pins, as shown on Figure 5.6-5.

The IFBA pin requirements shown in Figure 5.6-5 are based on nominal IFBA linear B^{10} loading of 1.57 mg-B^{10}/inch (1.0X). For higher IFBA loadings up to 2.35 mg-B^{10}/inch (1.5X), the required number of IFBA pins per assembly may be reduced by the ratio of the increased B^{10} loading to the nominal 1.57 mg-B^{10}/inch loading. A full length IFBA is 168 inches long, and a part length IFBA is greater than 120 inches long.

CATEGORY 5:

Fuel in Category 5 shall meet a least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 1.70 w/o; or
2) a minimum burnup as shown on Figure 5.6-6.

CATEGORY 6:

Fuel in Category 6 shall meet at least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 1.70 w/o; or
2) a minimum burnup as shown on Figure 5.6-7.

CATEGORY 7:

Fuel in Category 7 shall meet a least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 1.65 w/o; or
2) a minimum burnup as shown on Figure 5.6-8.

CATEGORY 8:

Fuel in Category 8 shall meet at least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 1.40 w/o; or
2) a minimum burnup as shown on Figure 5.6-9.

CATEGORY 9:

Fuel in Category 9 shall meet at least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 1.40 w/o; or
2) a minimum burnup as shown on Figure 5.6-10.

CATEGORY 10:

Fuel in Category 10 shall meet a least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 1.40 w/o; or
2) a minimum burnup as shown on Figure 5.6-11.

CATEGORY 11:

Fuel in Category 11 shall meet at least one of the following criteria:

1) an initial nominal enrichment of less than or equal to 1.20 w/o; or
2) a minimum burnup and decay time since shutdown as shown on Figure 5.6-12.

Data points for the curves presented in Figures 5.6-1 through 5.6-12 are presented in tables on the respective figures. Linear interpolation between table values may be used for intermediate points.

5.6.1.3 Region 1 racks may be used to store Category 1, 3, 4, 6, and 10 fuel. The fuel in Region 1 shall be stored in accordance with Figures 5.6-13, 5.6-15, and 5.6-16.

Empty water cells may be substituted for fuel assemblies in all cases.

Empty water cells may be used to store non-fissile items provided that: the cells are not face-adjacent to a cell storing a fuel assembly, or an evaluation has been performed that supports storage of the non-fissile item.

5.6.1.4 Region 2 racks may be used to store Category 2, 5, 7, 8, 9, and 11 fuel. The fuel in Region 2 shall be stored in accordance with Figures 5.6-14, 5.6-17, 5.6-18, and 5.6-19.

Empty water cells may be substituted for fuel assemblies in all cases. Non-fissile items may be stored in empty water cells per the provisions of Specification 5.6.1.3.

5.6.1.5 Storage Configuration Interface Requirements. Fuel storage patterns used within Region 1 shall comply with the interface requirements shown in Figures 5.6-15 and 5.6-16. Fuel storage patterns used within Region 2 shall comply with the interface requirements shown in Figures 5.6-17 through 5.6-19. At the interface between Region 1 and Region 2 one row of empty water cells shall be maintained between the Regions. The empty water cell row can be positioned in either Region. Non-fissile items can be stored in the empty water cells per the provisions of Specification 5.6.1.3.

5.6.1.6 The minimum boron concentration of 2500 ppm specified by Specification 3.9.13, "Spent Fuel Pool Minimum Boron Concentration" bounds the boron concentration of 2200 ppm required for the most limiting fuel misloading and also assures that the rack K_{eff} limit in Specification 5.6.1.1.a will not be violated.

5.6.1.7 The new fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 when filled with aqueous foam moderation (low density water). This requirement shall be met by limiting the fuel assembly nominal enrichments to 5.0 w/o or less.

b. A nominal 21 inches center to center distance between fuel assemblies.

5.6.1.8 The In-containment fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water. This requirement shall be met by satisfying at least one of the following criteria:

1) a maximum initial fuel assembly nominal enrichment to 4.5 w/o or less; or,

2) a minimum number of Integral Fuel Burnable Absorbers (IFBA), as a function of initial nominal assembly enrichment, as shown on Figure 5.6-20, or a K_{inf} of less than or equal to 1.484. The fuel assembly K_{inf} shall be based on a unit assembly configuration (infinite in the lateral and axial extent) in the reactor core geometry, assuming unborated water at 68°F.

The IFBA pin requirements shown in Figure 5.6-20 are based on a nominal IFBA linear B^{10} loading of 1.57 mg-B^{10}/inch. For higher IFBA linear B^{10} loadings, the required number of IFBA pins per assembly may be reduced by the ratio of the increased B^{10} loading to the nominal 1.57 mg-B^{10}/inch loading.

b. A nominal 16 inches center to center distance between fuel assemblies.

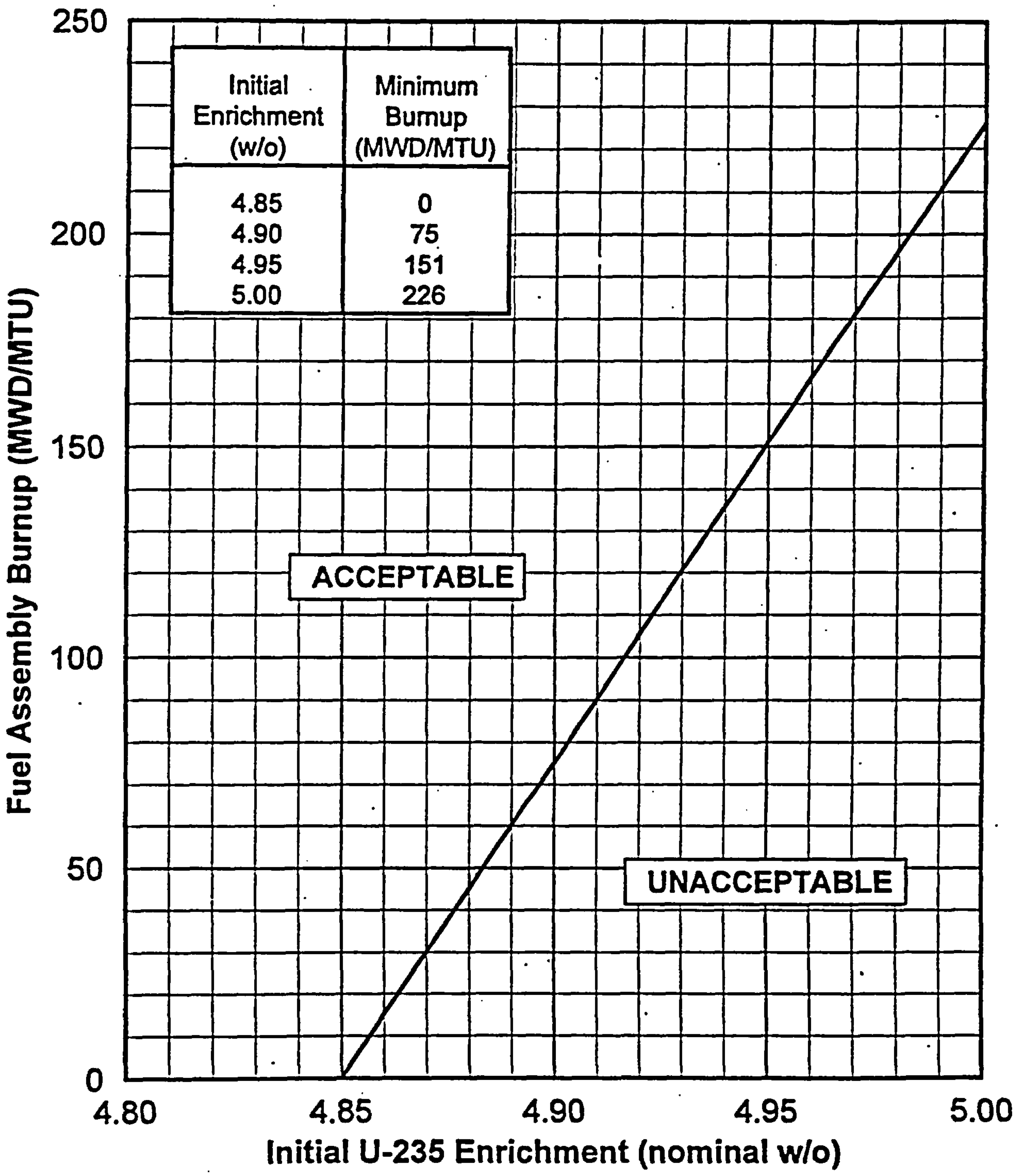
DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 62 feet-6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1969 fuel assemblies.

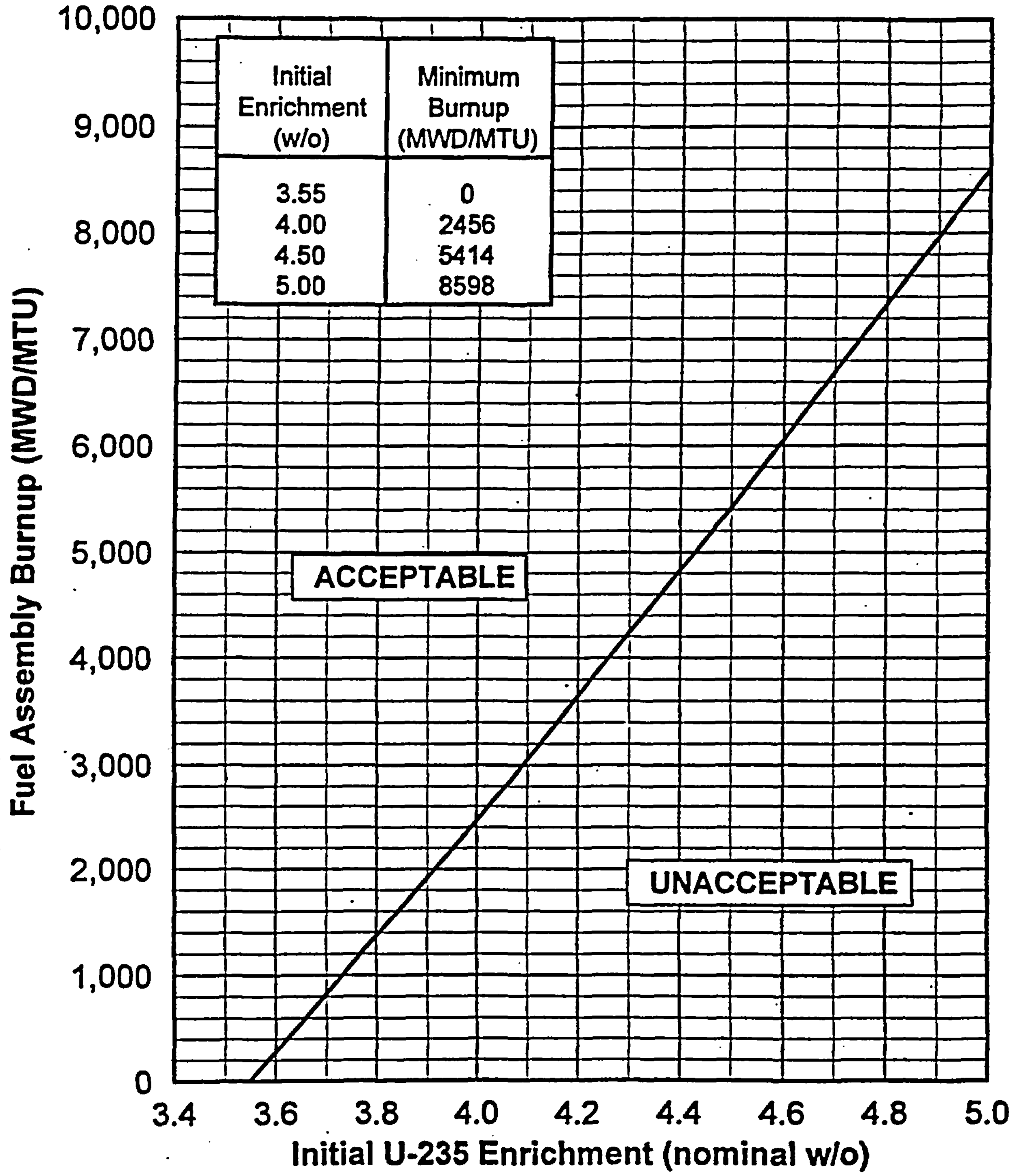
Minimum Burnup for Category 2 Fuel
Region 2



Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)
4.85	0
4.90	75
4.95	151
5.00	226

Figure 5.6-1

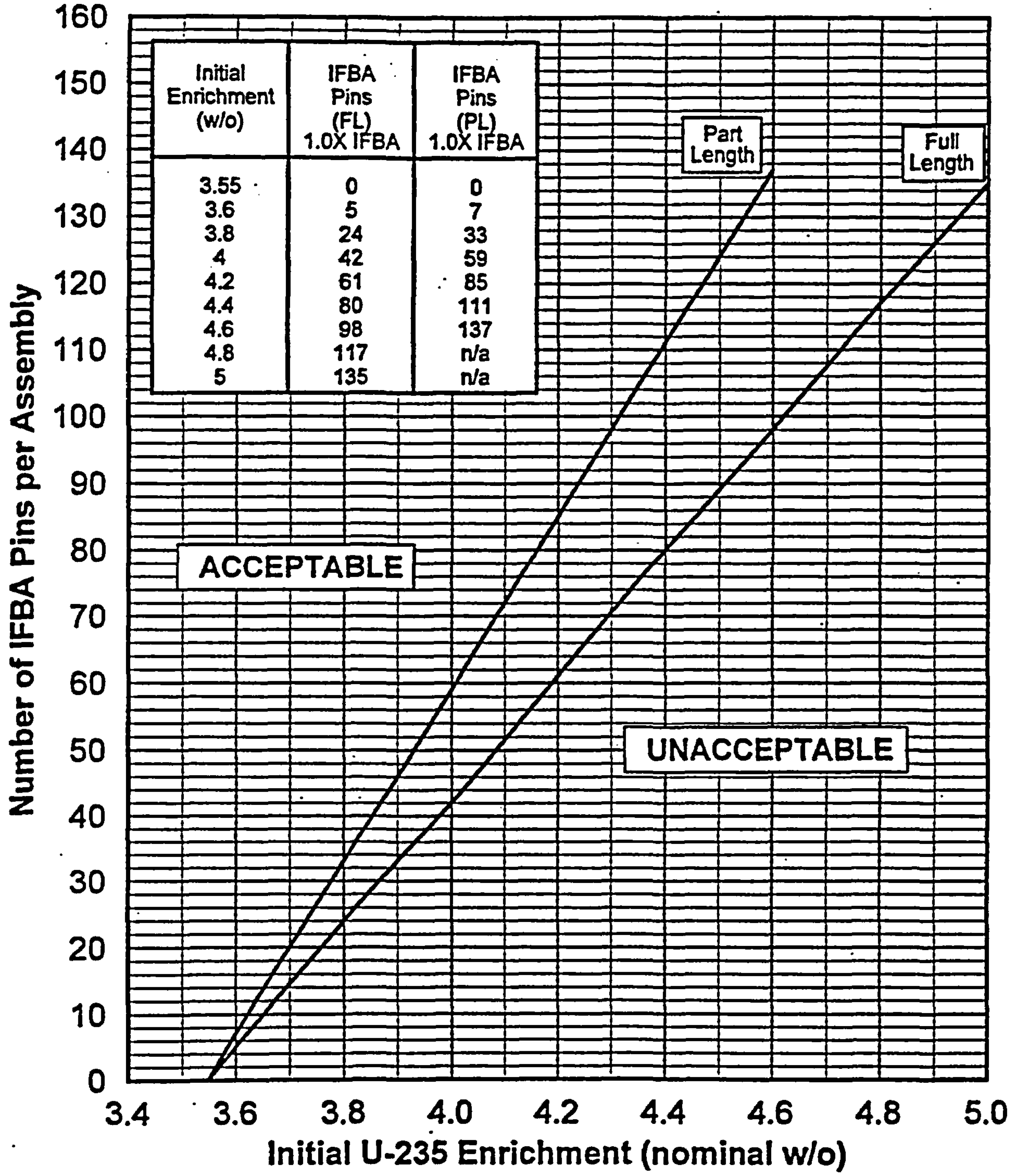
Minimum Burnup for Category 3 Fuel
Region 1



Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)
3.55	0
4.00	2456
4.50	5414
5.00	8598

Figure 5.6-2

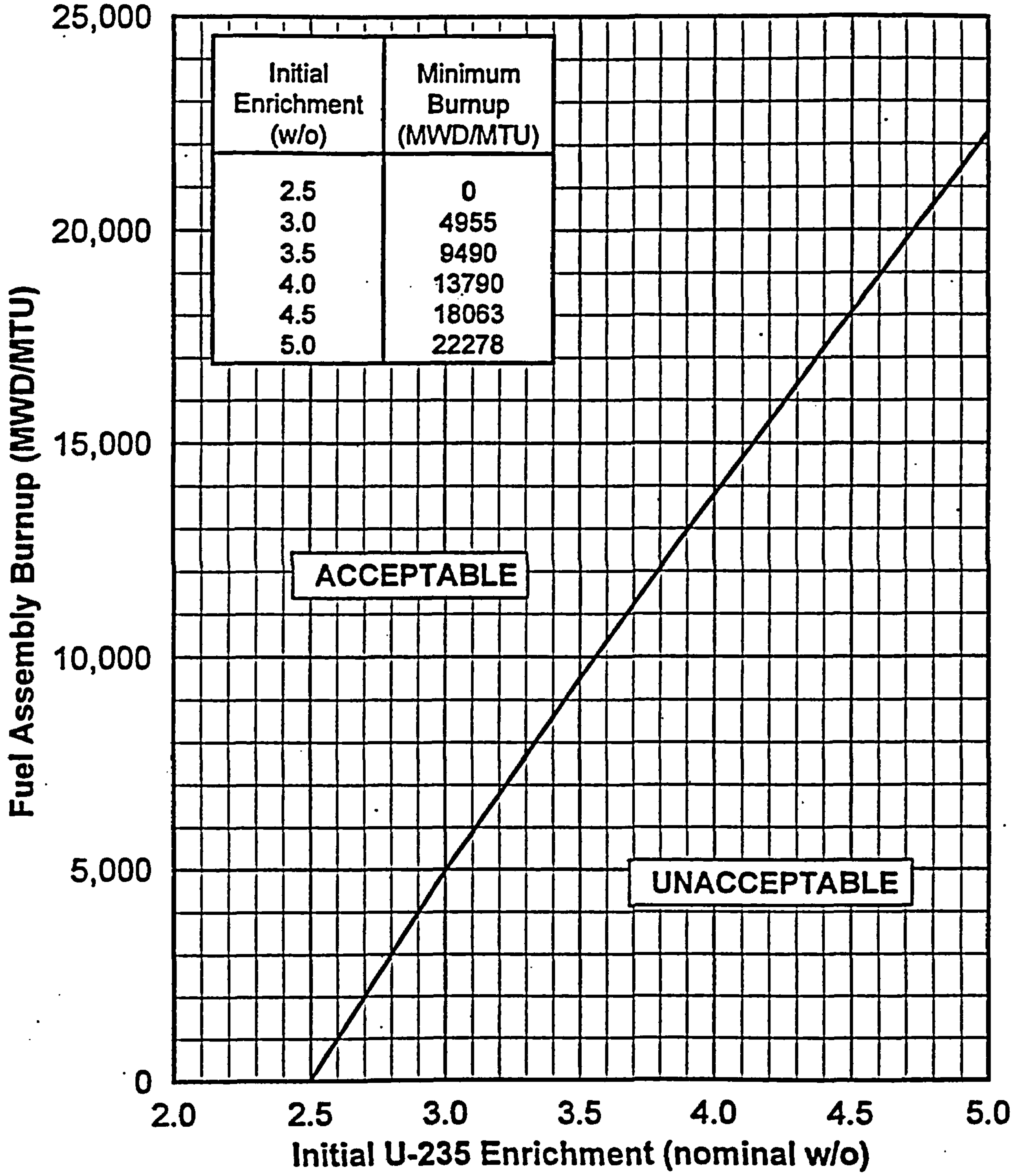
Minimum IFBA for Category 3 Fuel
Region 1



Initial Enrichment (w/o)	IFBA Pins (FL) 1.0X IFBA	IFBA Pins (PL) 1.0X IFBA
3.55	0	0
3.6	5	7
3.8	24	33
4	42	59
4.2	61	85
4.4	80	111
4.6	98	137
4.8	117	n/a
5	135	n/a

Figure 5.6-3

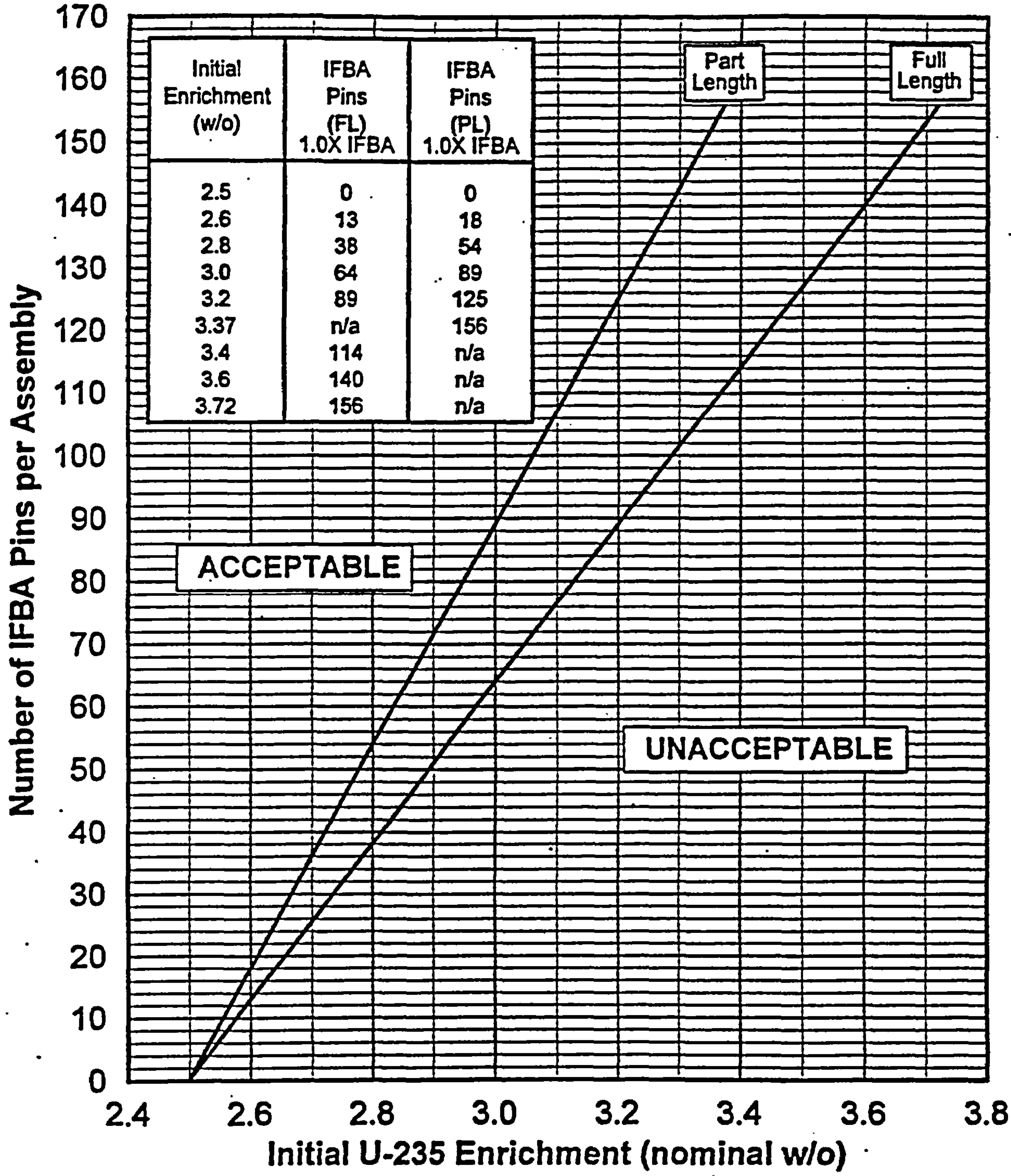
Minimum Burnup for Category 4 Fuel
Region 1



Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)
2.5	0
3.0	4955
3.5	9490
4.0	13790
4.5	18063
5.0	22278

Figure 5.6-4

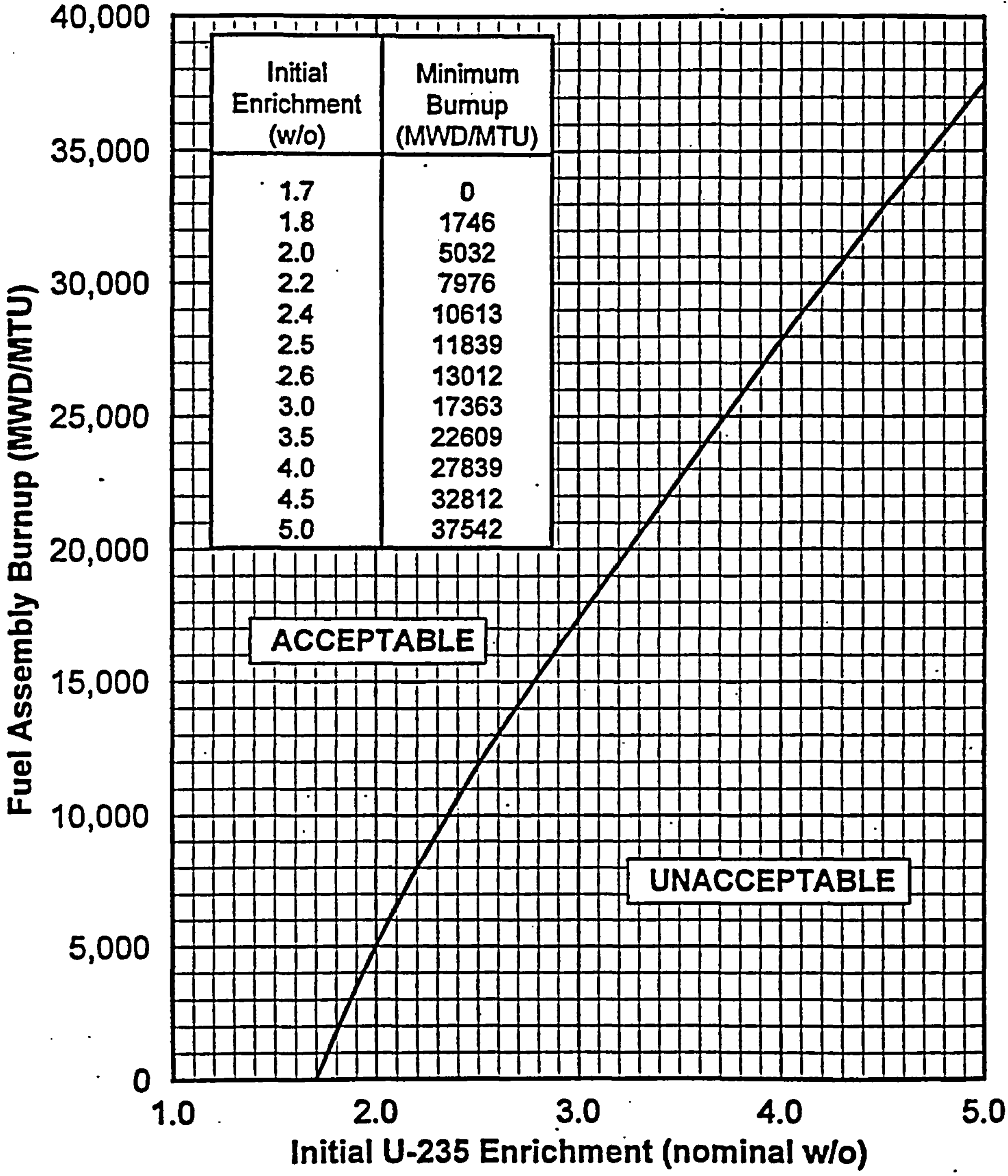
Minimum IFBA for Category 4 Fuel
Region 1



Initial Enrichment (w/o)	IFBA Pins (FL) 1.0X IFBA	IFBA Pins (PL) 1.0X IFBA
2.5	0	0
2.6	13	18
2.8	38	54
3.0	64	89
3.2	89	125
3.37	n/a	156
3.4	114	n/a
3.6	140	n/a
3.72	156	n/a

Figure 5.6-5

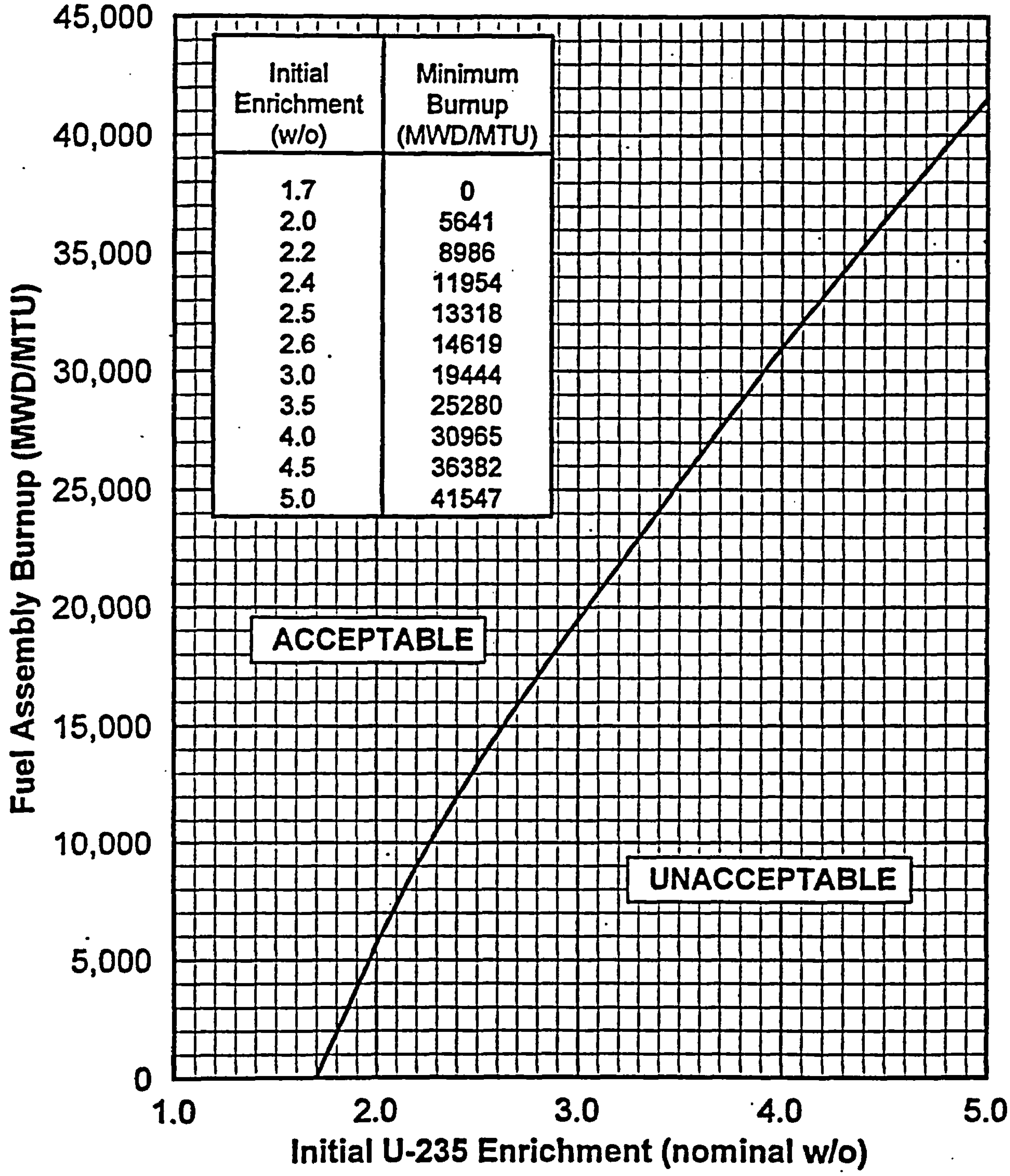
Minimum Burnup for Category 5 Fuel
Region 2



Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)
1.7	0
1.8	1746
2.0	5032
2.2	7976
2.4	10613
2.5	11839
2.6	13012
3.0	17363
3.5	22609
4.0	27839
4.5	32812
5.0	37542

Figure 5.6-6

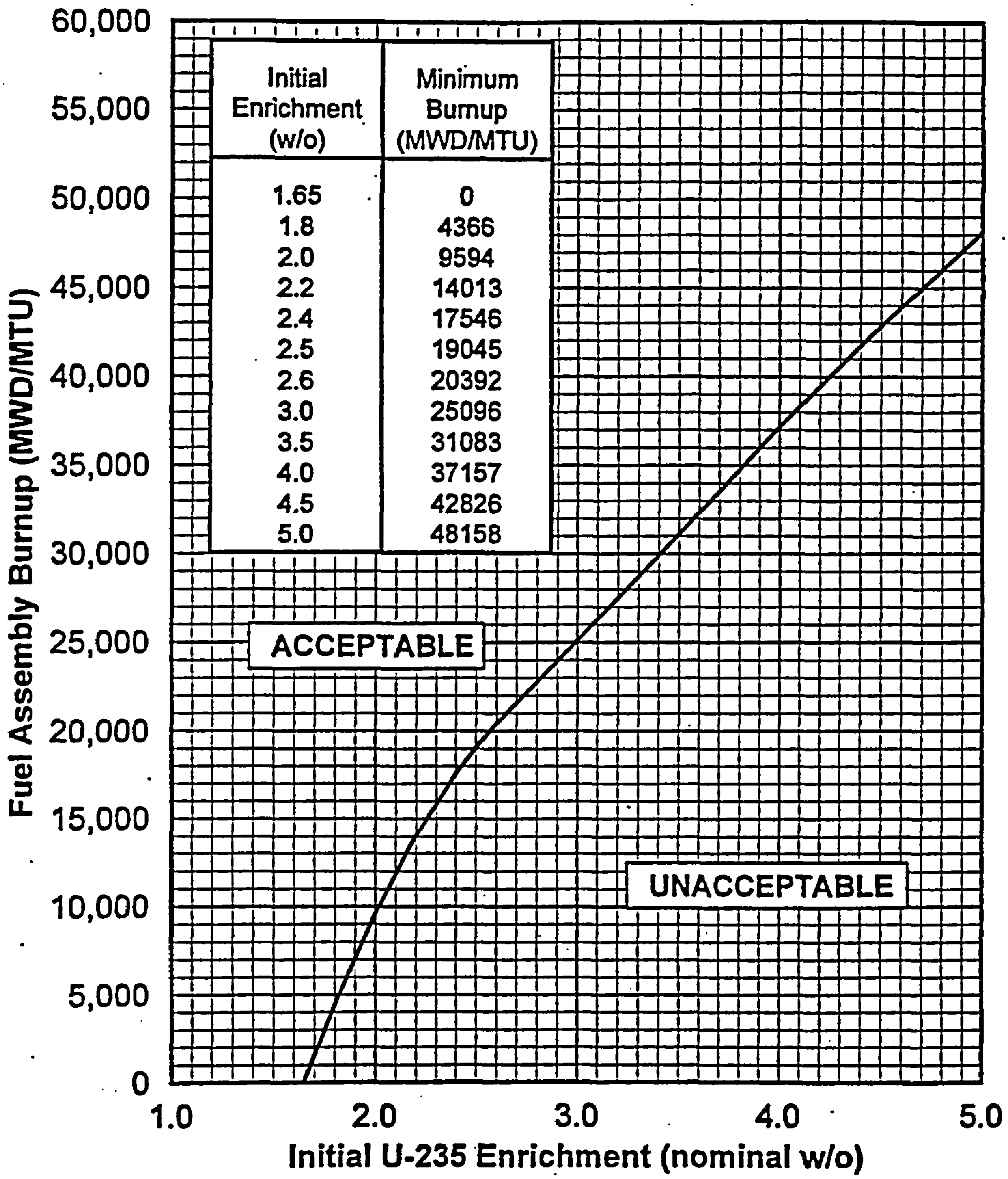
Minimum Burnup for Category 6 Fuel
Region 1



Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)
1.7	0
2.0	5641
2.2	8986
2.4	11954
2.5	13318
2.6	14619
3.0	19444
3.5	25280
4.0	30965
4.5	36382
5.0	41547

Figure 5.6-7

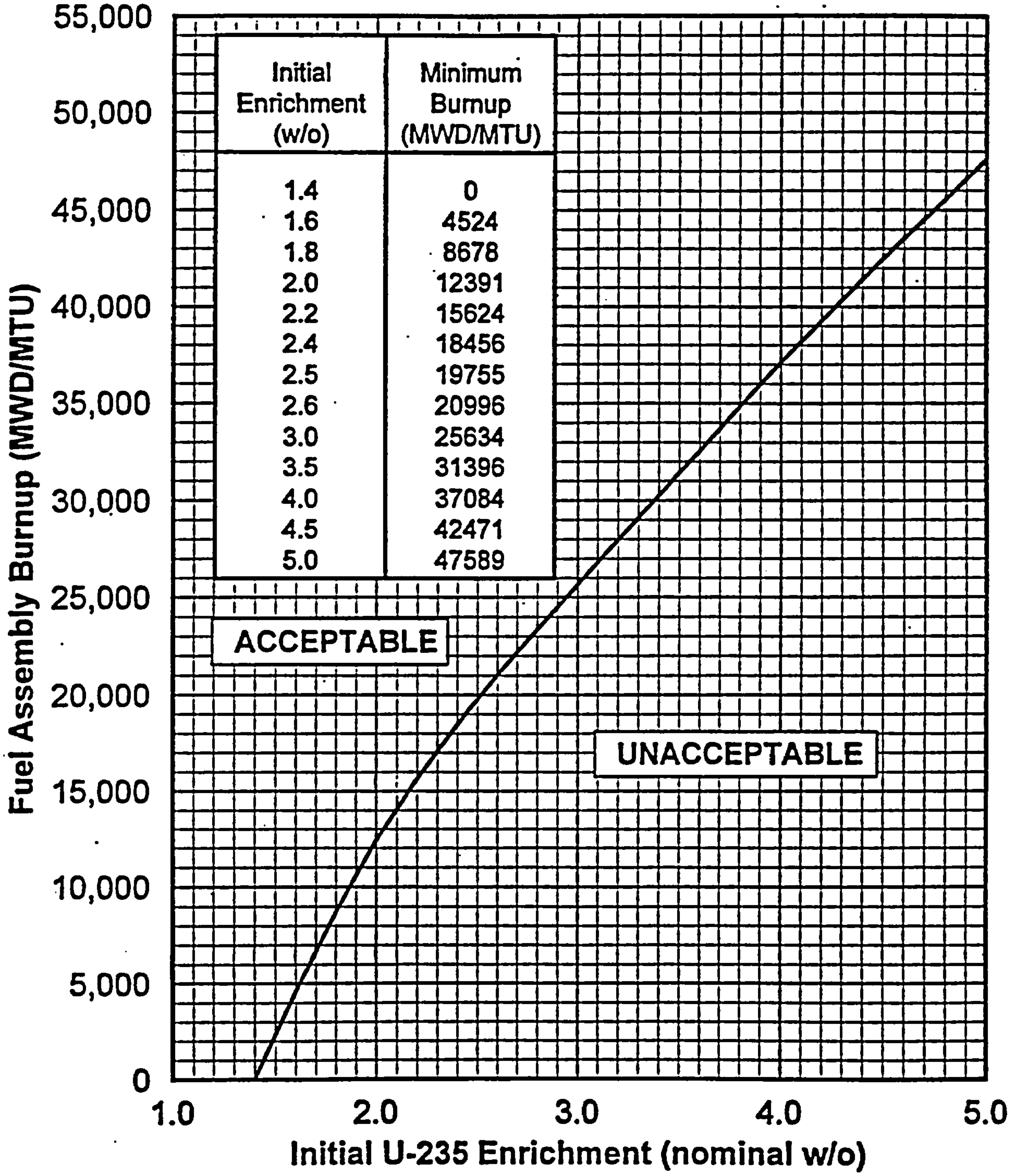
Minimum Burnup for Category 7 Fuel
Region 2



Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)
1.65	0
1.8	4366
2.0	9594
2.2	14013
2.4	17546
2.5	19045
2.6	20392
3.0	25096
3.5	31083
4.0	37157
4.5	42826
5.0	48158

Figure 5.6-8

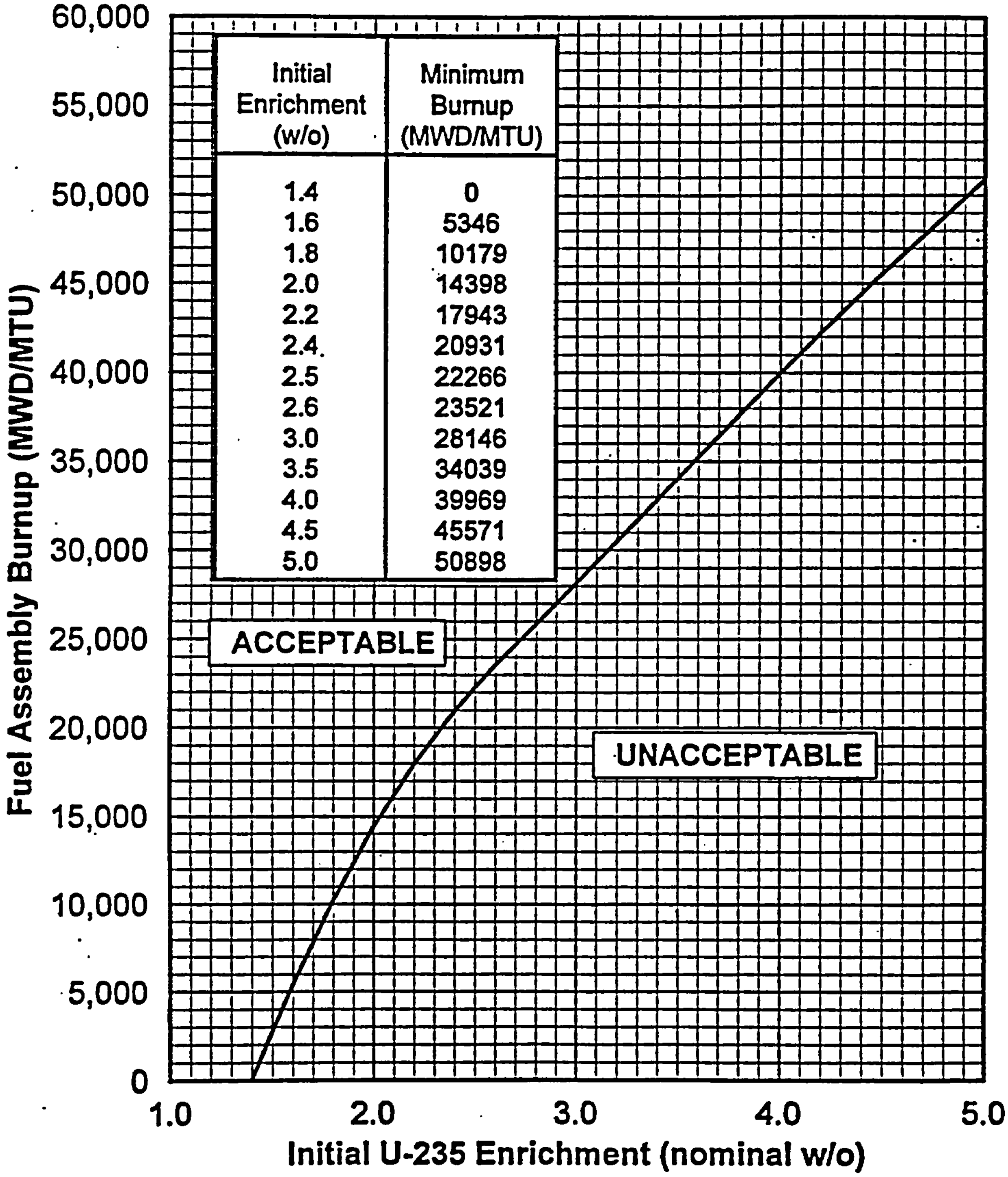
Minimum Burnup for Category 8 Fuel
Region 2



Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)
1.4	0
1.6	4524
1.8	8678
2.0	12391
2.2	15624
2.4	18456
2.5	19755
2.6	20996
3.0	25634
3.5	31396
4.0	37084
4.5	42471
5.0	47589

Figure 5.6-9

Minimum Burnup for Category 9 Fuel
Region 2



Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)
1.4	0
1.6	5346
1.8	10179
2.0	14398
2.2	17943
2.4	20931
2.5	22266
2.6	23521
3.0	28146
3.5	34039
4.0	39969
4.5	45571
5.0	50898

Figure 5.6-10

Minimum Burnup for Category 10 Fuel
Region 1

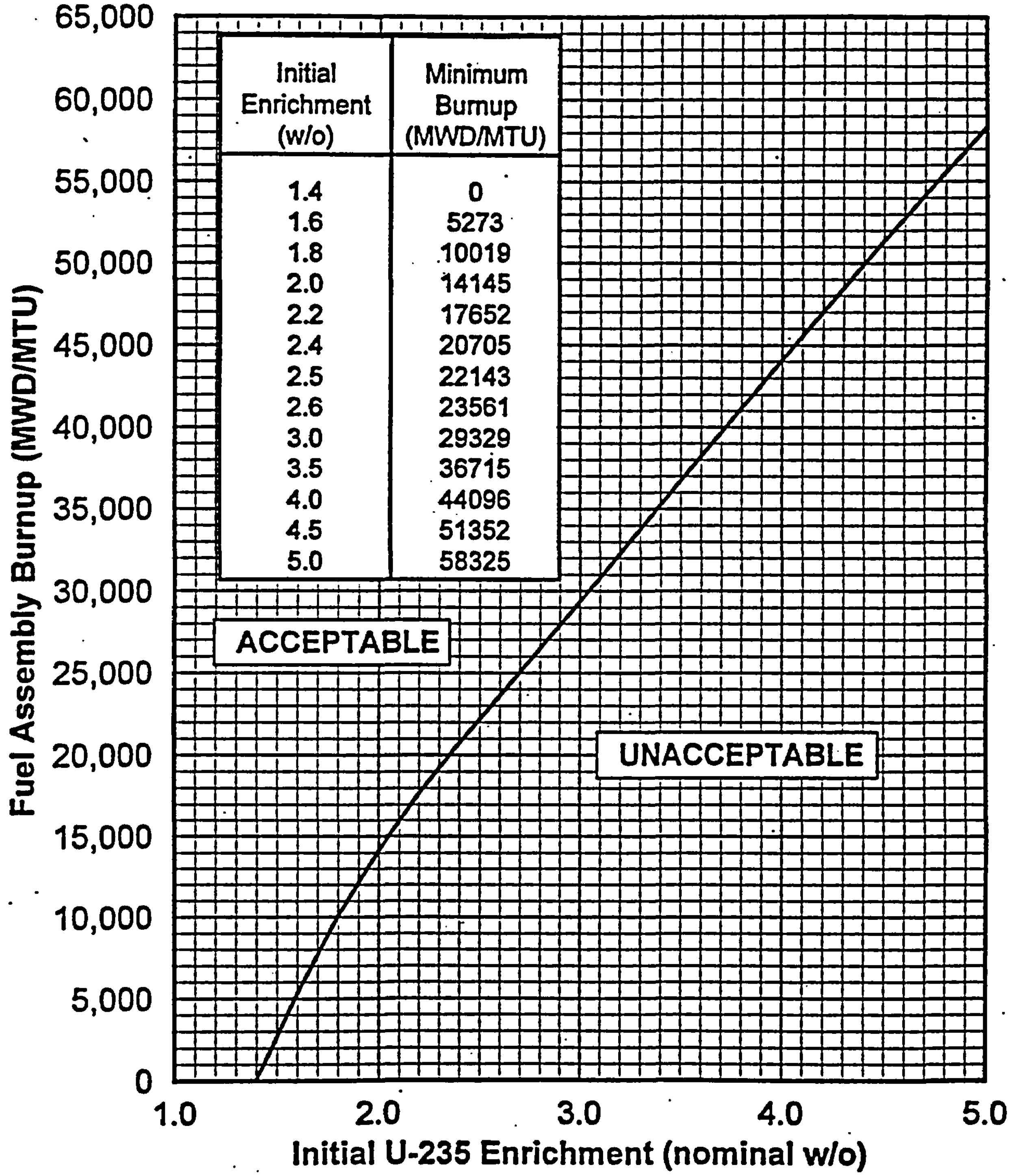
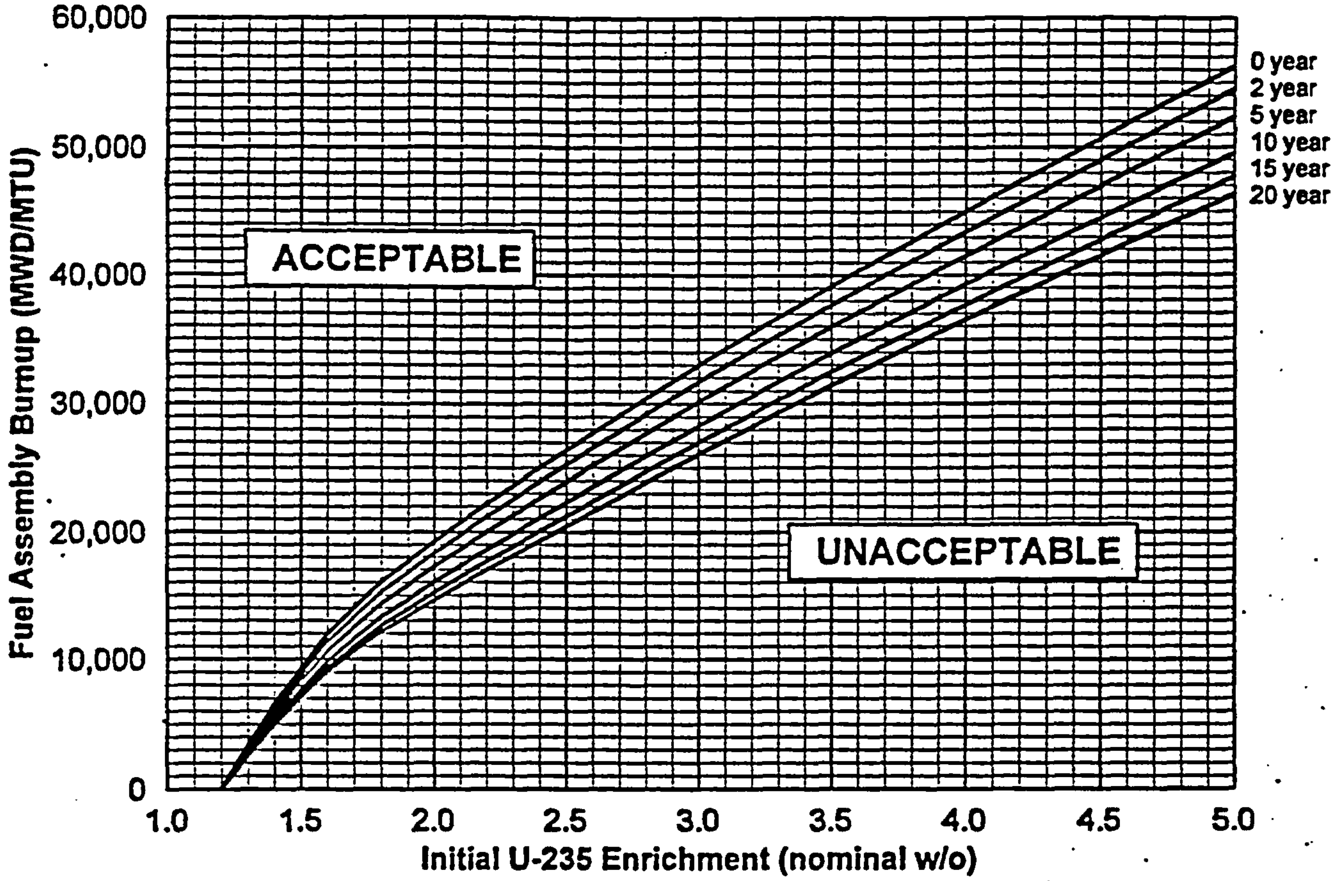


Figure 5.6-11

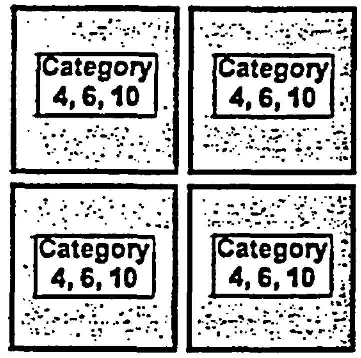
Minimum Burnup for Category 11 Fuel
Region 2



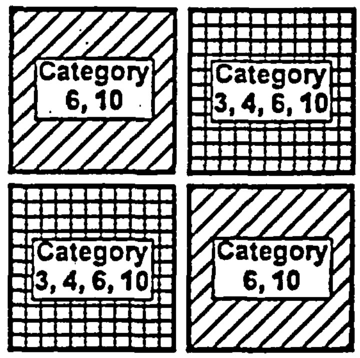
Initial Enrichment (w/o)	Minimum Burnup (MWD/MTU)					
	0 year	2 year	5 year	10 year	15 year	20 year
1.2	0	0	0	0	0	0
1.4	6533	6188	5807	5384	5113	4937
1.6	11912	11298	10613	9843	9347	9018
1.8	16021	15221	14313	13282	12606	12149
2.0	19209	18285	17221	15992	15177	14611
2.2	22150	21130	19939	18543	17609	16952
2.4	24988	23888	22590	21050	20012	19276
2.6	27732	26563	25173	23507	22377	21574
2.8	30389	29156	27686	25912	24700	23838
3.0	32967	31669	30126	28260	26973	26057
3.2	35474	34107	32495	30550	29192	28226
3.4	37919	36481	34803	32787	31363	30348
3.6	40314	38807	37064	34981	33493	32432
3.8	42669	41098	39291	37138	35591	34485
4.0	44995	43368	41499	39267	37664	36514
4.2	47300	45629	43697	41374	39718	38525
4.4	49582	47878	45885	43459	41753	40518
4.6	51838	50112	48056	45519	43765	42489
4.8	54065	52325	50208	47550	45753	44437
5.0	56259	54513	52336	49551	47713	46358

Figure 5.6-12

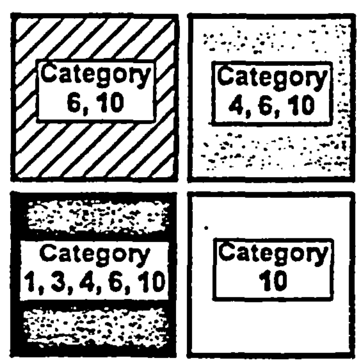
Region 1 All Cell



Region 1 Checkerboard #1



Region 1 Checkerboard #2

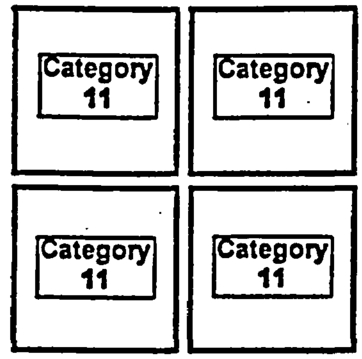


Note: These configurations can be rotated (90°, 180°, 270°) provided that configuration interface requirements are satisfied.

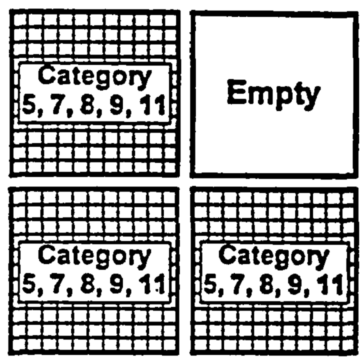
Allowable Fuel Categories for Region 1 Configurations

Figure 5.6-13

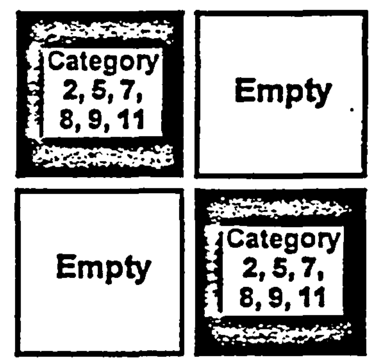
Region 2 All Cell



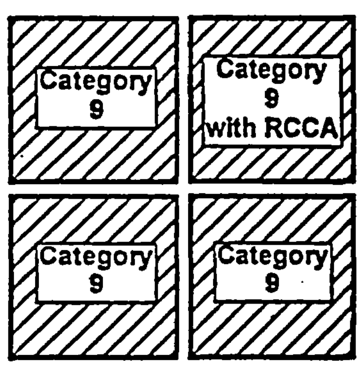
Region 2: 3-of-4 Storage



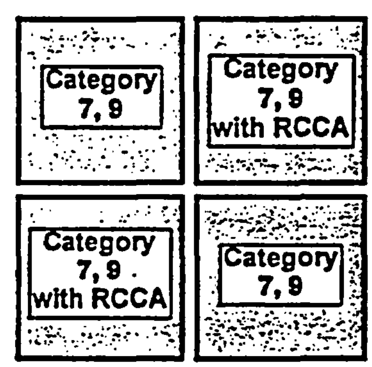
Region 2: 2-of-4 Storage



Region 2 RCCA Checkerboard #1



Region 2 RCCA Checkerboard #2



Note 1: Category 8 and 9 fuel can be substituted for any and all Category 11 fuel at the periphery of region 2. The periphery includes: cell locations next to the spent fuel pool wall, or cell locations separated from Region 1 by one row of empty cells.

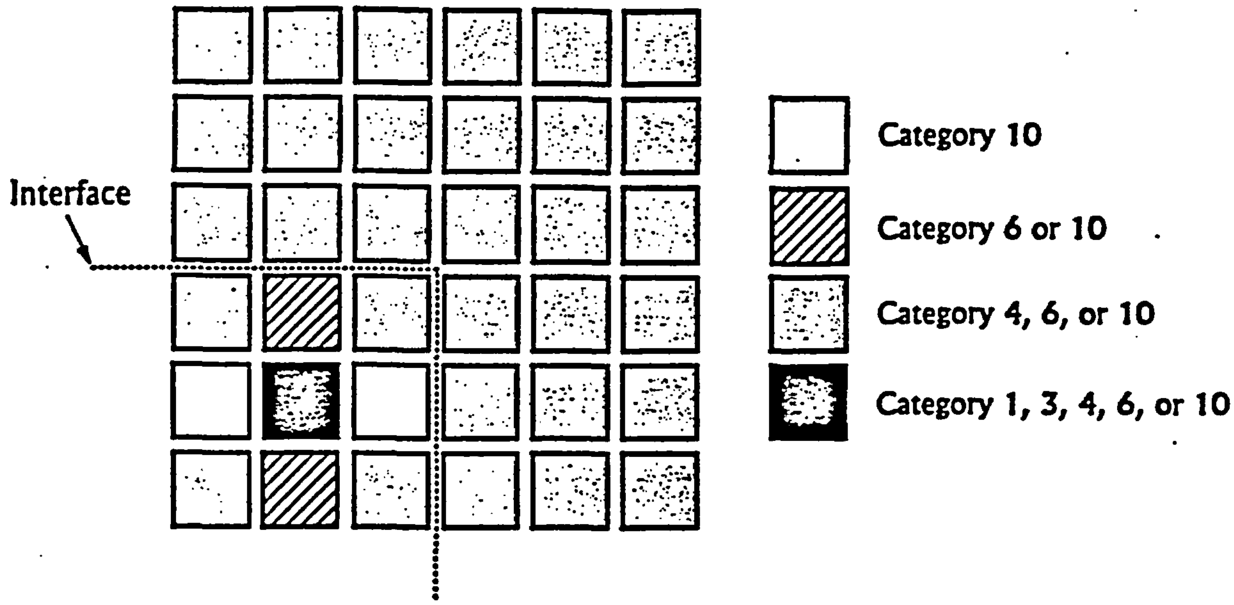
Note 2: See Technical Specification 5.6.1.4 for provisions for storing non-fissile items in empty cells.

Note 3: These configurations can be rotated (90°, 180°, 270°) provided that configuration interface requirements are satisfied.

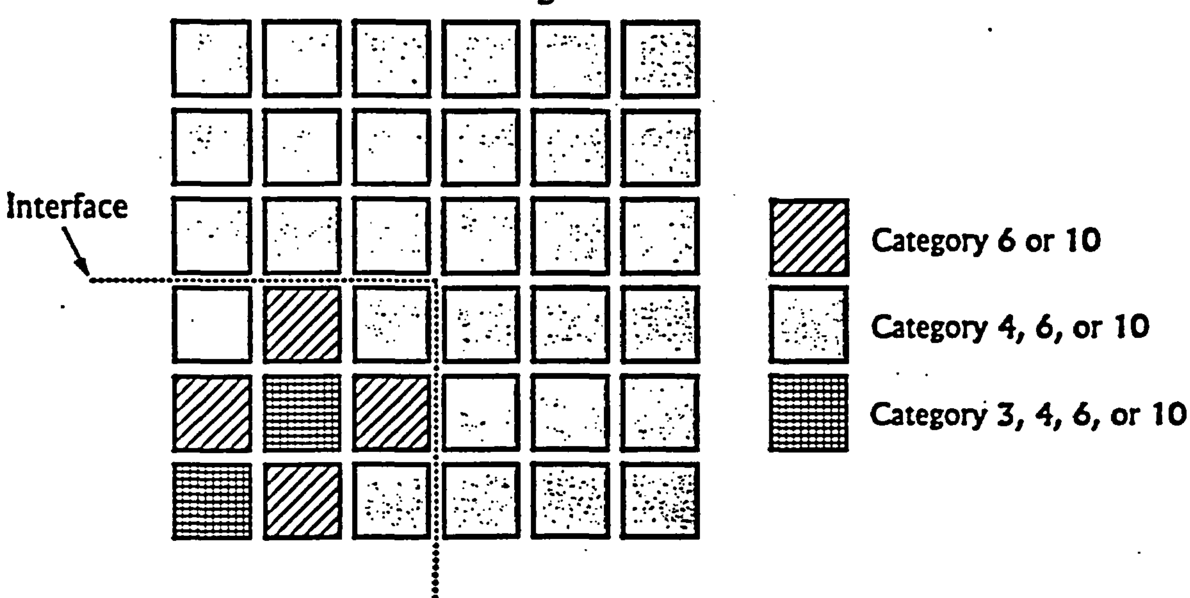
Allowable Fuel Categories for Region 2 Configurations

Figure 5.6-14

Region 1 Boundary Between All Cell Storage and Checkerboard #2



Region 1 Boundary Between All Cell Storage and Checkerboard #1

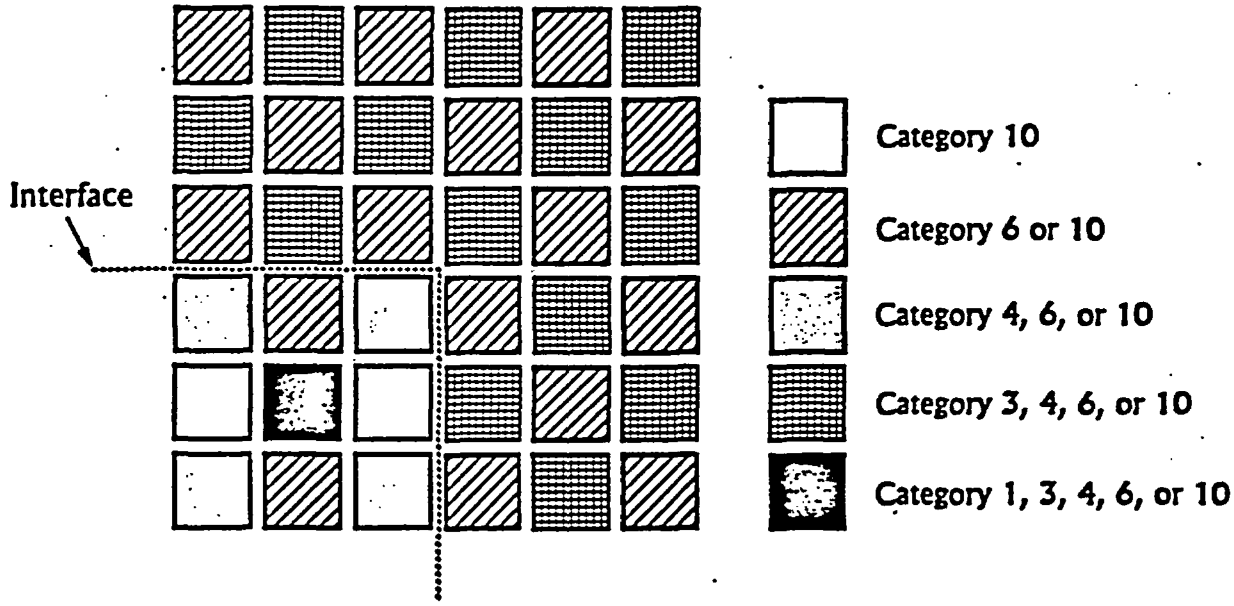


Note:

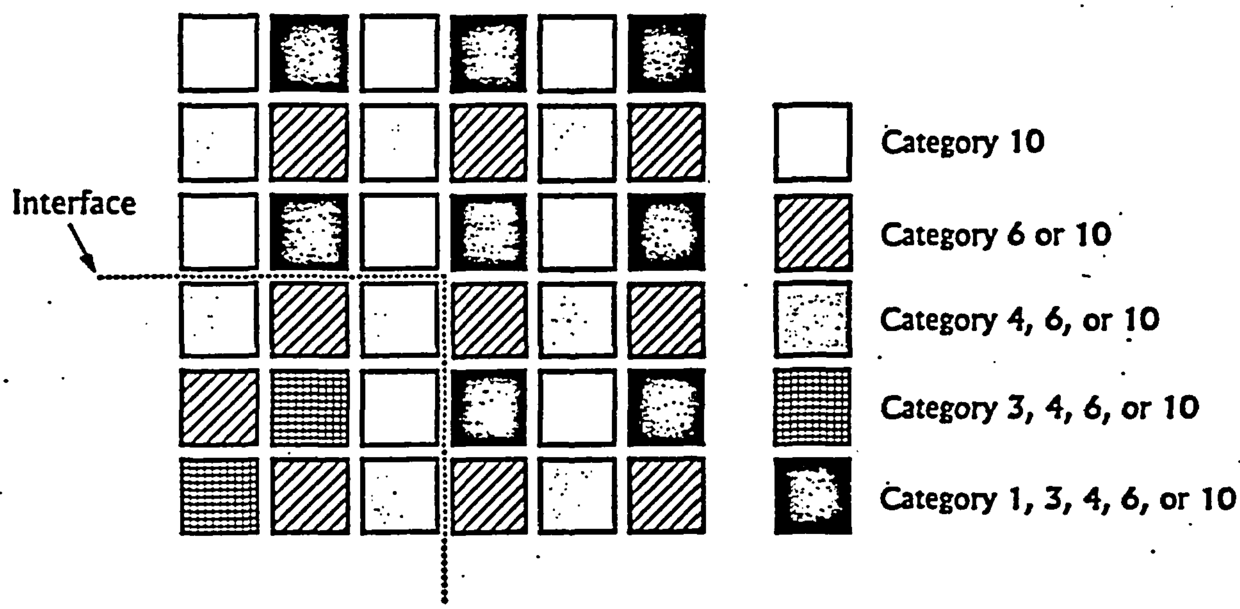
1. A row of empty cells can be used at the interface to seperate the configurations.
2. It is acceptable to replace an assembly with an empty cell.
3. See Specification 5.6.1.3 for provisions for storing non-fissile material in empty cells.

Figure 5.6-15

Region 1 Boundary Between Checkerboard #1 and Checkerboard #2



Region 1 Boundary Between Checkerboard #2 and Checkerboard #1

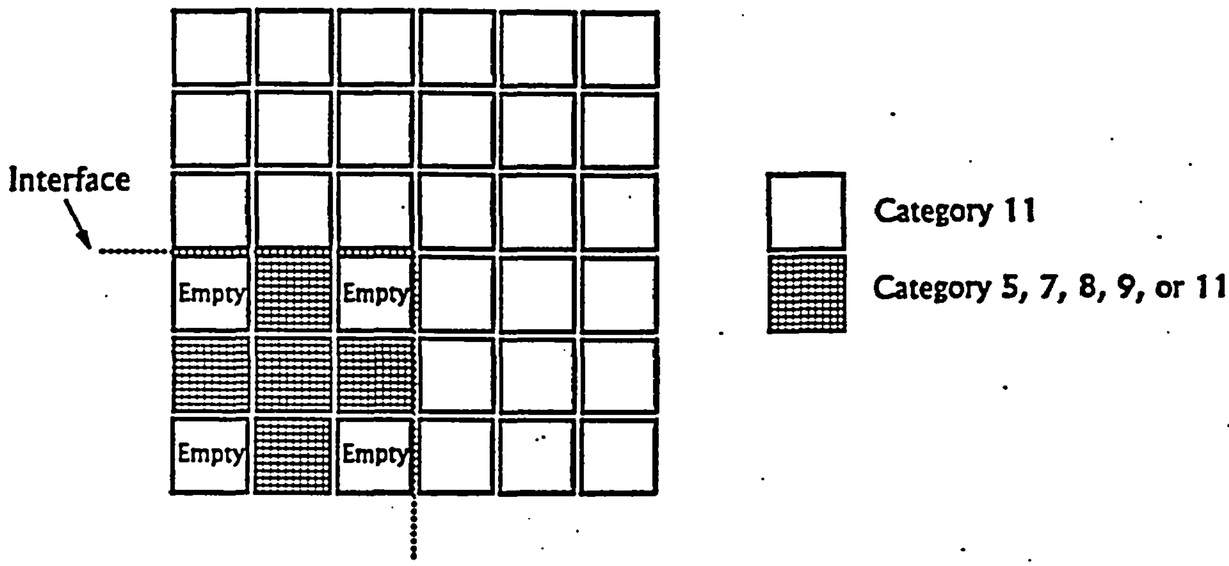


Note:

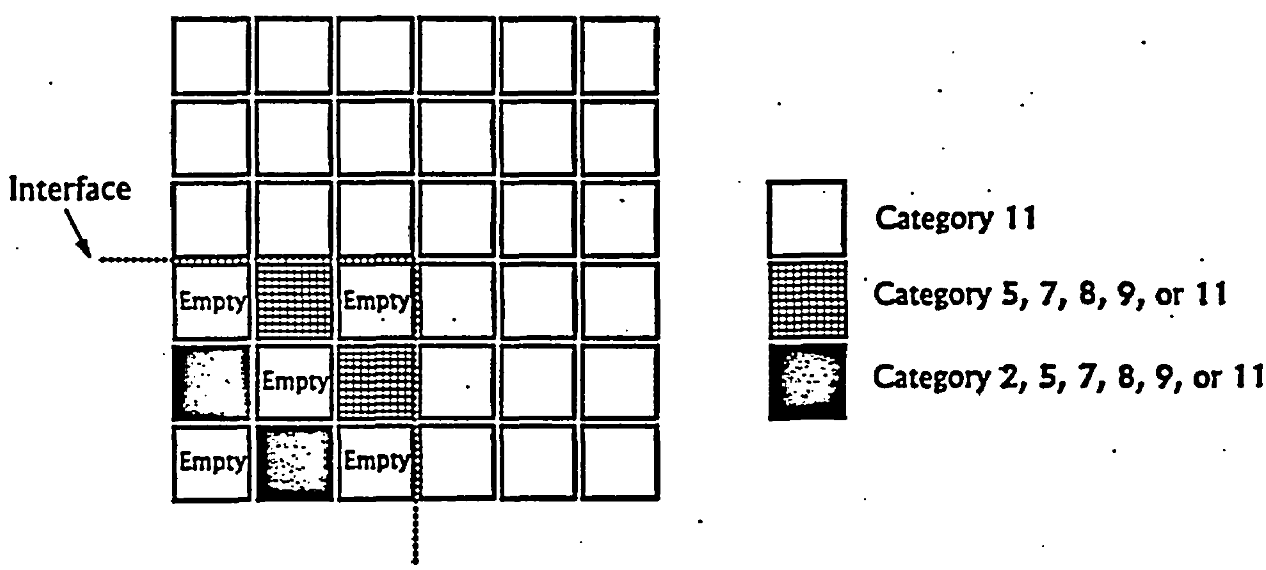
1. A row of empty cells can be used at the interface to seperate the configurations.
2. It is acceptable to replace an assembly with an empty cell.
3. See Specification 5.6.1.3 for provisions for storing non-fissile material in empty cells.

Figure 5.6-16

Region 2 Boundary Between All Cell Storage and 3-out-of-4 Storage



Region 2 Boundary Between All Cell Storage and 2-out-of-4 Storage

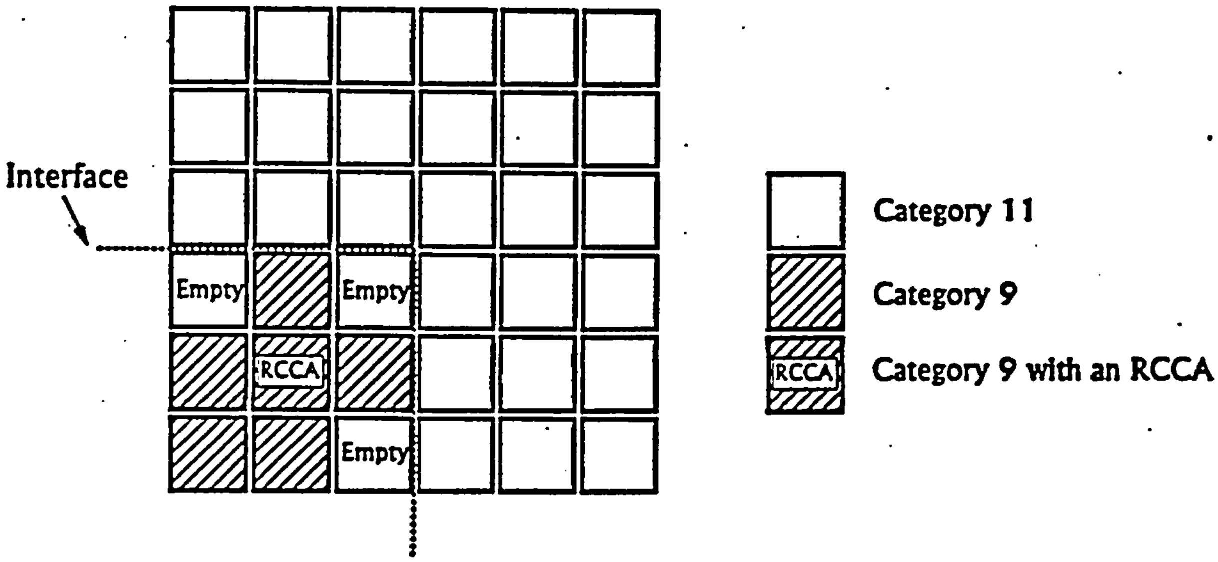


Note:

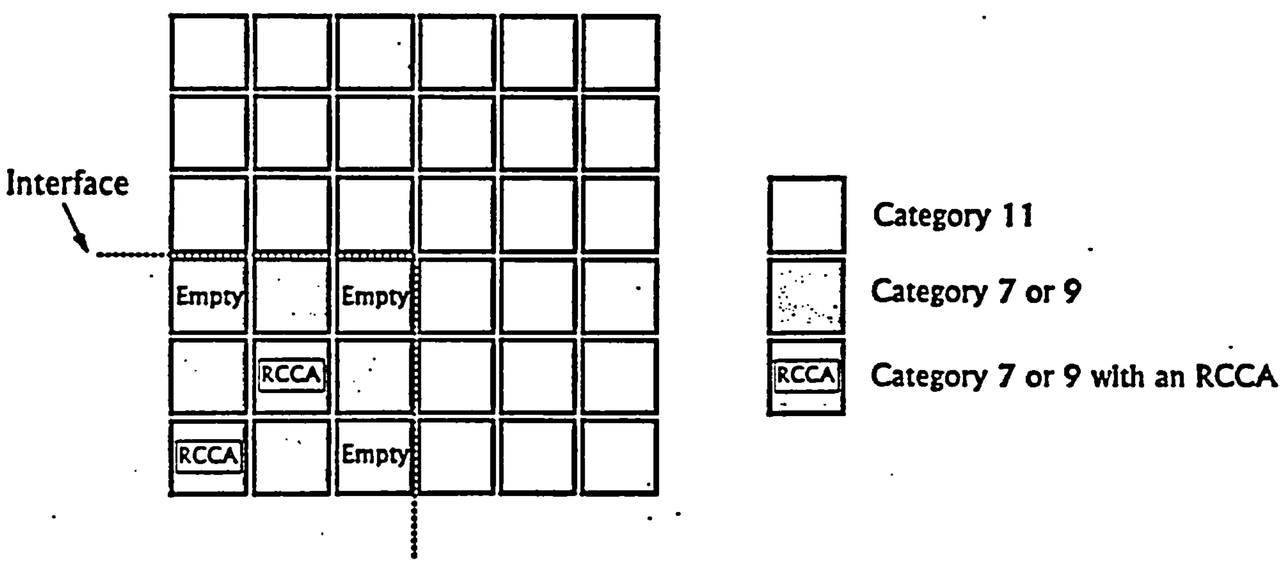
1. A row of empty cells can be used at the interface to seperate the configurations.
2. It is acceptable to replace an assembly with an empty cell.
3. See Specification 5.6.1.4 for provisions for storing non-fissile material in empty cells.

Figure 5.6-17

Region 2 Boundary Between All Cell Storage and RCCA Checkerboard #1



Region 2 Boundary Between All Cell Storage and RCCA Checkerboard #2

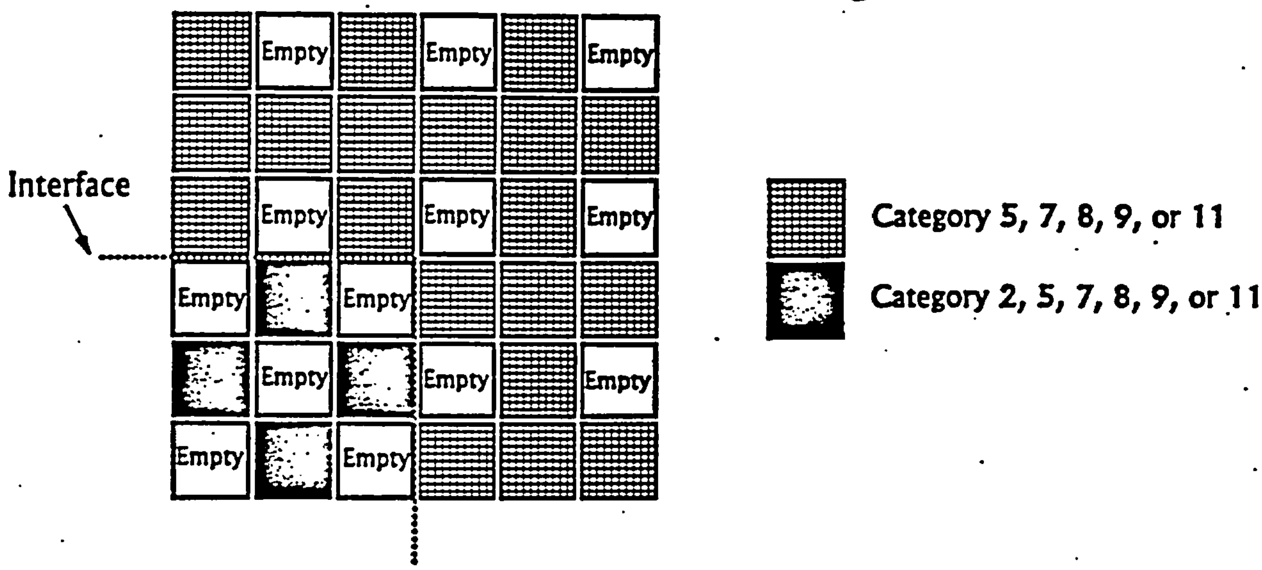


Note:

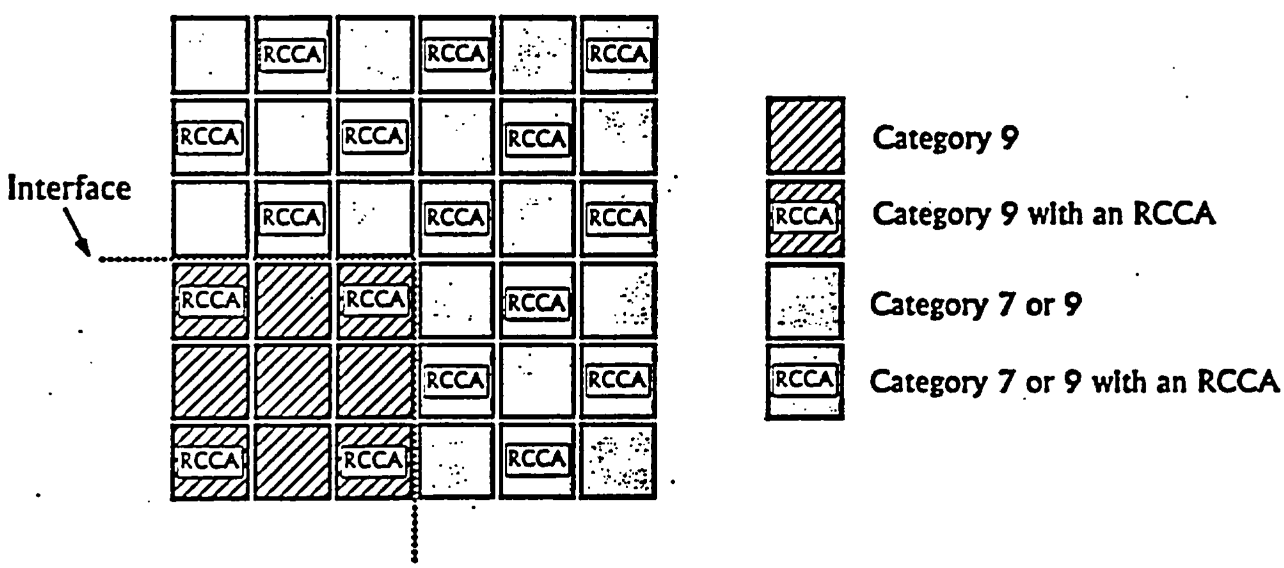
1. A row of empty cells can be used at the interface to seperate the configurations.
2. It is acceptable to replace an assembly with an empty cell.
3. See Specification 5.6.1.4 for provisions for storing non-fissile material in empty cells.

Figure 5.6-18

Region 2 Boundary Between 2-out-of-4 and 3-out-of-4 Storage



Region 2 Boundary Between RCCA Checkerboard Storage Patterns



Note:

1. A row of empty cells can be used at the interface to seperate the configurations.
2. It is acceptable to replace an assembly with an empty cell.
3. See Specification 5.6.1.4 for provisions for storing non-fissile material in empty cells.

Figure 5.6-19

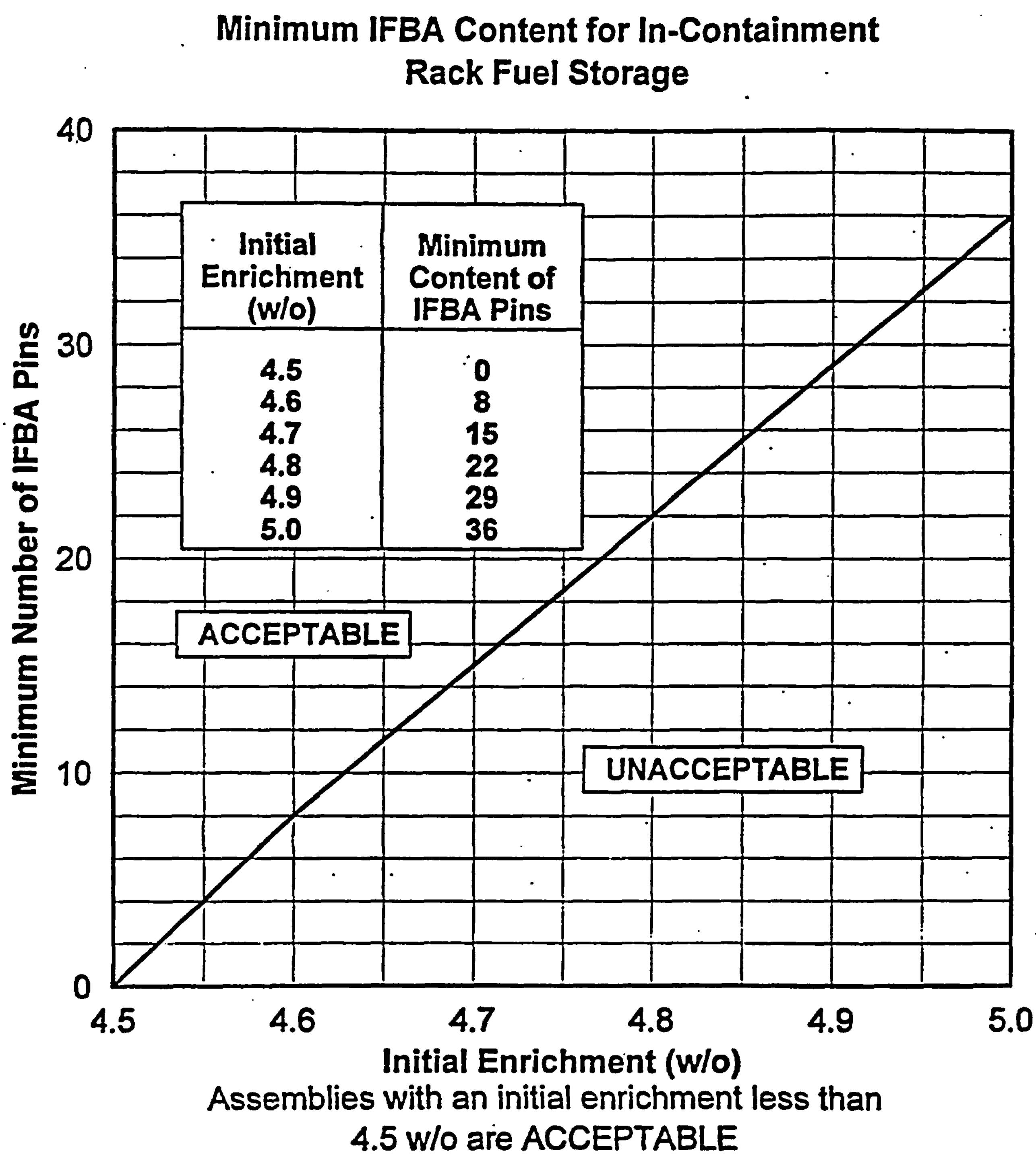

Minimum IFBA Content for In-Containment
Rack Fuel Storage
40
30
20
10
0
4.5
4.6
4.7
4.8
4.9
5.0
Minimum Number of IFBA Pins
Initial Enrichment (w/o)
Minimum Content of IFBA Pins
4.5 0
4.6 8
4.7 15
4.8 22
4.9 29
5.0 36
ACCEPTABLE
UNACCEPTABLE
Initial Enrichment (w/o)
Assemblies with an initial enrichment less than
4.5 w/o are ACCEPTABLE

Figure 5.6-20

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components of the reactor coolant system are designed and shall be maintained within limits addressed in the Component Cyclic and Transient Limit Program as required by specification 6.8.3f.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or the plant manager's designee shall approve, prior to implementation, each proposed test and experiment that affects nuclear safety and is not described in the UFSAR, and each modification to systems or equipment that affects nuclear safety.

6.1.2 The shift manager shall be responsible for the control room command function. During any absence of the shift manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function for that unit. During any absence of the shift manager from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function for that unit.

6.2.1 Offsite and Onsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be 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, job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the UFSAR and/or the Operations Quality Assurance Plan.

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff, carry out radiation protection functions, 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

The unit staff organization shall include the following:

a. A total of three non-licensed operators for the two units is required in all conditions. At least one of the required non-licensed operators shall be assigned to each unit. When a unit is operating in MODES 1, 2, 3, or 4, two non-licensed operators are required to be assigned to that unit.

b. The shift crew composition may be one less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specifications 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.

(continued)

6.2.2 Unit Staff (continued)

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

d. Not Used.

e. The individual to whom the shift managers directly report shall hold an SRO license.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the shift manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift (Generic Letter 86-04). This position may also be filled by the shift manager or an individual with an SRO license provided that person meets the qualifications specified by the Commission Policy Statement.

6.2.3 Not Used

6.2.4 Not Used

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, as described in the Operations Quality Assurance Plan.

6.4 Not Used

6.5 Not Used

6.6 Not Used

6.7 Not Used

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Quality Assurance Program for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. Programs and Manuals specified in Specification 6.8.3.

6.8.2 Not Used

6.8.3 The following programs and manuals shall be established, implemented, and maintained:

a. 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 systems include containment spray, safety injection, containment hydrogen monitoring, and primary sampling. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements; and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. Not Used

c. Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables;

2) Identification of the procedures used to measure the values of the critical variables;

(continued)

6.8.3.c (continued)

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage;

4) Procedures for the recording and management of data;

5) Procedures defining corrective actions for all off-control point chemistry conditions; and

6) A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

d. Not used

e. Not Used

f. Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9.1 cyclic/transient plant conditions to assure that the components are maintained within the design limits.

g. Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program shall be contained in the 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:

1) 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;

(continued)

6.8.3.g (continued)

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY from the radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to 10 CFR 50, Appendix I;

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

6) Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to 10 CFR 50, Appendix I;

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the following:

   a) For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   b) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ;

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas at or beyond the SITE BOUNDARY conforming to 10 CFR 50, Appendix I;

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas at or beyond the SITE BOUNDARY conforming to 10 CFR 50, Appendix I; and

(continued)

6.8.3.g (continued)

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR 190.

h. Not Used

i. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all based on applicable ASTM Standards. The purpose of the program is to establish the following:

1) Acceptability of new fuel oil prior to addition to the diesel generator fuel oil storage tanks by determining that the fuel oil has:

   a. an API gravity or absolute specific gravity within limits,

   b. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   c. a clear and bright appearance with proper color or a water and sediment content within limits;

2) Within 31 days following addition of new fuel oil to the diesel generator fuel oil storage tanks, verify that the properties of the new fuel oil, other than those addressed in 6.8.3.i.1 above, are within limits for ASTM 2D fuel oil; and

3) Total particulate concentration of fuel oil is $\leq$ 10 mg/l when tested every 31 days.

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

j. Containment Leakage Rate Testing Program

A program shall be established to implement 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 Nuclear Energy Institute (NEI) topical report NEI 94-01 Revision 2-A, dated October 2008.

(continued)

6.8.3.j (continued)

Peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA), P_a is 41.2 psig.

The maximum allowable containment leakage rate, L_a, is 0.3 percent of containment air weight per day.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit start-up following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq .75\ L_a$ as-left and $\leq 1.0\ L_a$ as-found for Type A tests.

2) Air lock testing acceptance criteria for the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The provisions of Surveillance Requirement 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.
The provisions of Surveillance Requirement 4.0.3 apply to the Containment Leakage Rate Testing Program.

k. Configuration Risk Management Program (CRMP)

A program to calculate risk-informed completion time in accordance with NEI 06-09, "Risk-Managed Technical Specifications (RMTS) Guidelines, Rev. 0". The CRMP may be used for calculating a risk-informed completion time only in Mode 1 and Mode 2. In accordance with NEI 06-09, the completion time determined using the CRMP shall not be more than 30 days.

(continued)

6.8.3 (continued)

l. Containment Post-Tensioning System Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measures prior to initial operations. The Containment Post-Tensioning System Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL, of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC.

The provisions of SR 4.0.3 are applicable to the Containment Post-Tensioning System Surveillance Program inspection frequencies with the exception of the surveillance interval extension allowed per Surveillance Requirement 4.0.2.

m. 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 FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.8.3.m.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

n. Offsite Dose Calculation Manual (ODCM)

1) 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

(continued)

6.8.3.n (continued)

2) The ODCM shall also contain descriptions of the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radiological Effluent Release Report required by Specifications 6.9.1.3 and 6.9.1.4.

3) 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 changes together with the appropriate analyses or evaluations justifying the changes and

2. A determination that the changes 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 do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b) Shall become effective after approval of the plant manager.

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 to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (month and year) the change was implemented.

o. 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.

6.8.3.o (continued)

1. Structural integrity performance criterion. All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 (3ΔP) 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. Accident induced leakage is not to exceed 1 gpm total for all four SGs in one unit.

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.

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.

6.8.3.o (continued)

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 nondestructive 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.

p. Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, which includes the following:

1) Actions to restore battery cells discovered with float voltage < 2.13 V;

2) Actions to equalize and test battery cells found with electrolyte level below the top of the plates;

3) Actions to verify that the remaining cells are > 2.07 V when a cell or cells are found to be < 2.13 V; AND

4) Actions to ensure that specific gravity readings are taken prior to each discharge test.

q. 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 Makeup and Cleanup Filtration System, 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:

1. The definition of the CRE and the CRE boundary.

2. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

6.8.3.q (continued)

3. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

   The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

   1) C.1.2 -- No peer reviews are required to be performed.

4. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by two trains of the Control Room Makeup and Cleanup Filtration System, operating at the flow rate required by the Surveillance Requirement 4.7.7.c.3, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

5. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph 3. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

6. The provisions of SR 3.0.2 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 3 and 4, respectively.

6.8.3.r 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 operations are met.

1) The Surveillance Frequency Control Program shall contain a list of frequencies of those surveillance requirements for which the frequency is controlled by the program.

2) 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.

   STP takes the following exception to NEI 04-10, “Risk-Informed Method for Control of Surveillance Frequencies,” Revision 1:

   a. STP will use the Independent Decisionmaking Panel (IDP) described in the applications approved by the NRC for the Graded Quality Assurance Program and the Exemption from Certain Special Treatment Requirements, augmented by the Surveillance Test Coordinator and Subject Matter Expert(s), to perform the IDP function.

3) The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

6.9.1 The following reports shall be submitted in accordance with 10 CFR 50.4.

6.9.1.1 Not Used

6.9.1.2 Not Used

6.9.1.3 Annual Radiological Environmental Operating Report

NOTE

A single submittal may be made for the South Texas Project. The submittal should combine sections that are common to both units.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 15 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM) and 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and 1V.C.

(continued)

6.9.1.4 Radioactive Effluent Release Report

NOTE

A single submittal may be made for the South Texas Project, which 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 by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents, and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program, and be in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.I.

6.9.1.5 Not Used

6.9.1.6 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle or prior to any remaining portion of a reload cycle. The core operating limits shall be documented in the COLR for the following:

1. Safety limits for thermal power, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) for Specification 2.1,
2. Limiting Safety System Settings for Reactor Coolant Flow-Low Loop design flow, Overtemperature ΔT, and Overpower ΔT setpoint parameter values for Specification 2.2,
3. SHUTDOWN MARGIN limits for Specification 3/4.1.1.1,
4. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3/4.1.1.3,
5. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
6. Control Bank Insertion Limits for Specification 3/4.1.3.6,
7. Axial Flux Difference limits and target band for Specification 3/4.2.1,

(continued)

6.9.1.6a (Continued)

8. Heat Flux Hot Channel Factor, K(Z), F_{xy} Exclusion Zones, Power Factor Multiplier, F_{xy}^{RTP}, and $F_Q(Z)$ manufacturing and measurement uncertainties for Specification 3/4.2.2,

9. Nuclear Enthalpy Rise Hot Channel Factor, Power Factor Multiplier, and $F_{\Delta H}^{N}$ measurement uncertainties for Specification 3/4.2.3, and

10. DNB related parameters for Reactor Coolant System T_{avg} Pressurizer Pressure, and the Minimum Measured Reactor Coolant System Flow for Specification 3/4.2.5.

The COLR shall be maintained available in the Control Room.

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. WCAP 9272-P-A, "Westinghouse reload safety evaluation methodology," July 1985 (W Proprietary).

(Methodology for Specification 3.1.1.1 - Shutdown Margin, Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 - DNB Parameters.)

2. WCAP 12942-P-A, "safety evaluation supporting a more negative eol Moderator temperature coefficient technical specification for the south texas project electric generating station units 1 and 2."

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

3. WCAP 8745-P-A, "Design Basis for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary Class 2).

(Methodology for Specification 2.1 - Safety Limits and 2.2 - Limiting Safety System Settings)

4. WCAP 8385, "power distribution and load following procedures topical report," September, 1974 (W Proprietary)

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control) and 3.2.2 – Heat Flux Hot Channel Factor (F_{xy} Exclusion Zones))

(continued)

6.9.1.6b (Continued)

5. Westinghouse Letter NS-TMA-2198, T.M. Anderson (Westinghouse) to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 – Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control). Approved by NRC Supplement No. 4 to NUREG-0422, January 1981, Docket Nos. 50-369 and 50-370.)

6. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

7. WCAP-10266-P-A, Rev. 2, WCAP-11524-NP-A Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Kabadi, J.N., et al., March 1987; including Addendum 1-A, "Power Shape Sensitivity Studies," December, 1987 and Addendum 2-A, "BASH methodology Improvements and Reliability Enhancements," May 1988.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor)

8.1 WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary)

8.2 WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™, July 2006 (W Proprietary)

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor)

9. Cameron Measurement Systems/Caldon Ultrasonics Engineering Report: ER-157(P-A) Rev. 8 and Rev. 8 Errata, "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System," May 2008.

(Methodology for operating at a RATED THERMAL POWER of 3,853 Mwt with LEFM CheckPlus System)

(continued)

6.9.1.6 (continued)

10. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (W Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

11. WCAP 12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994 (W Proprietary), including Addenda 1-A (January 2000) and 4 (September 2012)

(Methodology for uncertainties in Specification 3.2.2-Heat Flux Hot Channel Factor and 3.2.3-Nuclear Enthalpy Rise Hot Channel Factor)

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided to the NRC upon issuance for each reload cycle.

6.9.1.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.3.o, "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.

6.9.1.7 (continued)

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.

6.9.2 Not Used

6.10 Not Used

6.11 Not Used

6.12.1 Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of 20.1601(a), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is greater than 100 mrem/h but equal to or less than 1000 mrem/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., radiation protection technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with radiation levels equal to or less than 1000 mrem/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been determined and entry personnel are knowledgeable of them.

c. 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.

d. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   i. Be under the surveillance of an individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the RWP, or

   ii Be under surveillance by means of closed circuit television or equivalent, 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.

(continued)

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to individuals with radiation levels greater than 1000 mrem/h at 30 cm (12 in.) but less than 500 Rads in one hour at one meter from the radiation source or from any surface which the radiation penetrates shall be provided with locked or continuously guarded doors to prevent unauthorized entry. The keys to the doors shall be maintained under the administrative control of the shift manager on duty or radiation protection manager. Doors shall remain locked except during periods of access by individuals under an approved RWP. Prior to entry, individuals shall be informed of the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by individuals qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas, accessible to personnel, with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) but less than 500 Rads in one hour at one meter that are located within large areas, such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 Not Used

6.14 Not Used

APPENDIX B

TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

SOUTH TEXAS PROJECT

UNITS 1 AND 2

STP NUCLEAR OPERATING COMPANY, ET AL.

DOCKET NOS. 50-498 AND 50-499

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

MARCH 1989

SOUTH TEXAS PROJECT
UNITS 1 AND 2

ENVIRONMENTAL PROTECTION PLAN
(NONRADIOLOGICAL)

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1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of nonradiological environmental values during operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement - Operating Licensing Stage (FES-OL) and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters are regulated by way of the licensee's NPDES permit.

2.0 Environmental Protection Issues

In the FES-OL dated August 1986 (NUREG-1171), the staff considered the environmental impacts associated with the operation of the South Texas Project Unit Nos. 1 and 2. No aquatic/water quality, terrestrial, or noise issues were identified.

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in plant design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP*. Changes in plant design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

Before engaging in additional construction or operational activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measurable nonradiological environmental effects are confined to the on-site areas previously disturbed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

*This provision does not relieve the licensee of the requirements of 10 CFR 50.59.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level; or (3) a matter, not previously reviewed and evaluated in the documents specified in (1) of the Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in plant design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of the Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

3.2 Reporting Related to the NPDES Permit and State Certification

Changes to, or renewals of, the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The licensee shall notify the NRC of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES Permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, and local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. The following are examples: onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; fish kills; increase in nuisance organisms or conditions; and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decisions made by the U.S. Environmental Protection Agency and the state of Texas under the authority of the Clean Water Act for any requirements for aquatic monitoring.

4.2.2 Terrestrial Monitoring

No terrestrial monitoring is required.

4.2.3 Noise Monitoring

No noise monitoring is required.

4.2.4 Fog Monitoring

The licensee shall provide to the NRC the results of the fog monitoring program as described in Section 6.2.4.2 of Amendment 10 to the Environmental Report dated June 16, 1987. The report shall be provided within a reasonable time after completion and documentation of the results of the program.

5.0 Administrative Procedures

5.1. Review and Audit

The licensee shall provide for the review and audit of compliance with the EPP. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2. Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the station. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of others agencies.

5.3 Changes in Environmental Protection Plan

Requests for changes in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The period of the first report shall begin with the date of issuance of the Operating License for Unit 1, and the initial report shall be submitted prior to May 1 of the year following issuance of the Operating License for Unit 1.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 (if any) of this EPP for the report period, including a comparison with related preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends toward irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of mitigating action.

The Annual Environmental Operating Report shall also include:

(1) A list of EPP noncompliances and the corrective actions taken to remedy them.

(2) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental question.

(3) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing results shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall: (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact, and plant operating characteristics; (b) describe the probable cause of the event; (c) indicate the action taken to correct the reported event; (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems; and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided with a copy of such report at the same time it is submitted to the other agency.