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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

FACILITY OPERATING LICENSE

License No. NPF-48

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

   A. The application for license filed by the Duke Power Company acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (the licensees) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I; and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Catawba Nuclear Station, Unit 2 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-117 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 operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

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

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

*Duke Power Company is authorized to act as agent for the North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

G. The issuance of this 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 Facility Operating License No. NPF-48, 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;

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

2. Based on the foregoing findings and the July 26, 1985, and the November 21, 1985, affirmations by the Atomic Safety and Licensing Appeal Board of the Partial Initial Decisions issued by the Atomic Safety and Licensing Boards dated June 22, September 18, and November 27, 1984, regarding this facility and satisfaction of conditions therein imposed, Facility Operating License No. NPF-48 is hereby issued to the Duke Power Company, the North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (the licensees) to read as follows:

A. This license applies to the Catawba Nuclear Station, Unit 2, a pressurized water reactor and associated equipment (the facility) owned by the North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency and operated by Duke Power Company. The facility is located on the licensees' site in York County, South Carolina, on the shore of Lake Wylie approximately 6 miles north of Rock Hill, South Carolina, and is described in Duke Power Company's Final Safety Analysis Report, as supplemented and amended, and in its Environmental Report, as supplemented and amended;

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

(1) Duke Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in York County, South Carolina, in accordance with the procedures and limitations set forth in this license;

(2) North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency, pursuant to the Act and 10 CFR Part 50, to possess the facility at the designated location in York County, South Carolina, in accordance with the procedures and limitations set forth in this license;

(3) Duke Power Company, 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) Duke Power Company, 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) Duke Power Company, 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;

(6) Duke Power Company, 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; and

(7) Duke Power Company, 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 McGuire Nuclear Station, Units 1 and 2, and Oconee Nuclear Station, Units 1, 2, and 3.

C. This 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>

Duke Power Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100 percent power) in accordance with the conditions specified herein. Pending Commission approval, this license is restricted to power levels not to exceed 5 percent of full power (170 megawatts thermal).

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Initial Startup Test Program (Section 14, SER, SSER #3)*

Any changes to the Initial Test Program described in Section 14 of the FSAR 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.

(4) Antitrust Conditions

Duke Power Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(5) Inservice Inspection Program (Sections 5.2.4 and 6.6, SSER #2; Section 6.6, SSER #5)

Within six months of the date of this license, Duke Power Company shall submit the balance of the inservice inspection program as described in its letter dated January 8, 1985, for staff review and approval.

(6) Fire Protection Program (Section 9.5.1, SER, SSER #1, SSER #2, SSER #3, SSER #4, SSER #5)

(a) Duke Power Company shall maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to provisions b & c below.

(b) Duke Power Company may make no change to features of the approved fire protection program which would decrease the level of fire protection in the plant without prior approval of the Commission. To make such a change Duke Power Company must submit an application for license amendment pursuant to 10 CFR 50.90.

(c) Duke Power Company may make changes to features of the approved fire protection program which do not decrease the level of fire protection without prior Commission approval, provided:

(i) such changes do not otherwise involve a change in a license condition or technical specification or involve an unreviewed safety question (see 10 CFR 50.59).

*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.

(ii) such changes do not result in failure to complete the fire protection program approved by the Commission prior to license issuance.

Duke Power Company shall maintain, in an auditable form, a current record of all such changes including an analysis of the effects of the change on the fire protection program and shall make such records available to NRC inspectors upon request. All changes to the approved program made without prior Commission approval shall be reported to the Director of the Office of Nuclear Reactor Regulation together with the FSAR revisions required by 10 CFR 50.71(e).

(7) Turbine Missiles (Section 3.5.1.3, SER)

Duke Power Company shall submit for NRC staff approval by December 6, 1987, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities acceptable to the NRC staff or volumetrically inspect all low pressure turbine rotors within three years or by the second refueling outage, whichever is later, and thereafter every three years or every other refueling outage until a maintenance program is approved by the staff.

(8) Detailed Control Room Design Review, I.D.1 (Section 18.1, SER, SSER #2, SSER #5)

Duke Power Company shall correct all human engineering deficiencies according to the schedule contained in its letter dated March 28, 1984.

(9) Emergency Response Capabilities (Generic Letter 82-33, Supplement 1 to NUREG-0737).

(a) Regulatory Guide 1.97, Revision 2, Compliance (Section 7.5.2, SSER #4, SSER #5)

Prior to startup following the first refueling outage, Duke Power Company shall provide qualified accumulator discharge instrumentation.

(b) Safety Parameter Display System (SPDS) (Section 18.2, SSER #5)

Prior to startup following the first refueling outage, Duke Power Company shall add to the existing SPDS and have operational the following SPDS parameters: (a) residual heat removal flow, (b) containment isolation status, (c) stack radiation measurements, (d) primary coolant system hot leg temperature, and (e) steam generator or steamline radiation. The actual value of these and all other SPDS variables should be displayed for operator viewing in easily and rapidly accessible display formats.

(10) Anticipatory Reactor Trip, II.K.3.10 (Section 5.2.2, SER)

Prior to exceeding 70% power, Duke Power Company shall complete the described turbine trip tests to verify that PORVs will not be challenged when the anticipatory trip bypass is in effect.

(11) Steam Generator Tube Rupture (Section 15.4.4, SER, SSER #2)

Prior to startup following the first refueling outage of Catawba Unit 2, Duke Power Company shall submit for NRC staff review and approval an analysis which demonstrates that the steam generator single-tube rupture analysis presented in the FSAR is the most severe case with respect to the release of fission products and calculated doses. Consistent with the analytical assumptions, Duke Power Company shall propose any necessary changes to Appendix A to this license.

(12) Main Steam Line Break (MSLB) Outside Containment (Section 3.11, SSER #4, SSER #5)

Prior to startup following the first refueling outage, Duke Power Company: (a) shall provide the additional information identified in its November 15, 1985, letter, (b) shall provide any necessary information and clarification requested by the staff regarding the approval of the LOFTRAN Code used in the MSLB analysis and its resulting consequences, and (c) shall receive staff approval regarding the LOFTRAN Code used in the MSLB analysis and its resulting consequences.

(13) Transamerica Delaval, Inc. (TDI) Diesel Generators (Section 8.3.1, SSER #4, SSER #5)

Duke Power Company shall implement the TDI diesel requirements as specified in Attachment 1, and shall incorporate these requirements, within six months of the date of this license, into its maintenance and surveillance program. Attachment 1 is hereby incorporated into this license.

(14) Generic Letter 83-28 (Section 15.6, SSER #4, SSER #5)

Duke Power Company shall submit responses to and implement the guidance of Generic Letter 83-28 on a schedule which is consistent with that given in its November 2 and December 31, 1984, letters.

D. The facility requires exemptions from certain requirements of Appendices A and J to 10 CFR Part 50, as delineated below, and pursuant to evaluations contained in the referenced SER and SSERs. These include (a) partial exemption from General Design Criteria 16, 38, and 50 of Appendix A, with respect to the completion and testing of the ice condenser prior to the reactor coolant system temperature's exceeding 200°F (mode 4), in accordance with the Facility Technical Specifications the ice condenser is not required to be operable in Modes 5 and

6. (Section 6.2.1 of SSER #5), (b) partial exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J, the testing of containment airlocks at times when the containment integrity is not required (Section 6.2.6 of SSER #5), (c) exemption from the requirement of paragraph III.A.1(d) of Appendix J, insofar as it requires the venting and draining of lines for type A tests (Section 6.2.6 of SSER #5), and (d) partial exemption from the requirements of paragraph III.B of Appendix J, as it relates to bellows testing (Section 6.2.6 of the SER, and Section 6.2.6 of SSER #5). These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and certain special circumstances are present. These exemptions are, therefore, hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission. In addition, two exemptions were previously granted pursuant to 10 CFR 50.12. A partial exemption from those portions of General Design Criterion 4 of Appendix A to 10 CFR 50 which require protection of structures, systems and components against dynamic effects associated with postulated reactor coolant system pipe breaks was granted on April 23, 1985, for a period ending with the completion of the second refueling outage for Catawba Unit 2 or the adoption of the proposed rulemaking for modification of GDC-4 whichever occurs first. Furthermore, an exemption from the requirements of Appendix E, IV.F, insofar as they may require the active participation of all Crisis Management Center personnel for the Catawba Station emergency preparedness exercises (Section 13.3 of SSER #4), was granted on January 17, 1985, by the issuance of Facility Operating License No. NPF-35 for Catawba Nuclear Station, Unit 1.

E. Duke Power Company shall fully implement and maintain in effect all provisions of the Commission approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Catawba Nuclear Station Security Plan," with revisions submitted through December 16, 1985; "Catawba Nuclear Station Guard Training and Qualification Plan," with revisions submitted through September 30, 1985; and "Catawba Nuclear Station Safeguards Contingency Plan," with revisions submitted through October 23, 1985.

F. Reporting to the Commission

Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, Duke Power Company shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within twenty-four (24) hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 30 days in accordance with the procedures described in 10 CFR 50.73 (b), (c), and (e).

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

H. This license is effective as of the date of issuance and shall expire at midnight on February 24, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Attachment 1
2. Appendix A - Technical Specifications
3. Appendix B - Environmental Protection Plan
4. Appendix C - Antitrust Conditions

Date of Issuance: February 24, 1986

*SEE PREVIOUS PAGE FOR CONCURRENCES

with noted changes to SER 20.5 on exemption & no issuance till SER on exemption published → now published 2-24-86

PWR#4:DPWR-A	PWR#4:DPWR-A	PWR#4:DPWR-A	OELD	~~D:DPWR-A~~
KJabbour:kab	*MDuncan	*BJYoungblood	J. GRAY	TNovak
01/23/86 + 2/14/86	01/23/86	01/23/86	02/19 /86	02/24/86
DD:NRR	D:NRR	SP	PAS (subject to typos changed)	
DEisenhut	HDenton	IDinitz	WLambe	
02/ /86	02/24/86	02/20 /86	02/18 /86	

February 24, 1986

ATTACHMENT 1 TO LICENSE NPF-48

TDI DIESEL ENGINES REQUIREMENTS

Duke Power Company shall comply with the following requirements related to the TDI diesel engines for Catawba Unit 2.

1. Changes to the maintenance and surveillance program for the TDI diesel engines, as identified in Section 8.3.1.1.2(D) of SSER #5, shall be subject to the provisions of 10 CFR 50.59.

2. Connecting rod assemblies shall be subjected to the following inspections at each major engine disassembly (approximately every 5 years):

   ° The clearance between the link pin and the link rod should be examined. This dimension must be zero when the specified bolt torque is applied.

   ° The surfaces of the rack teeth should be inspected for signs of fretting. If fretting has occurred, it should be subject to an engineering evaluation for appropriate corrective action. The mating surfaces should also be examined to ensure that the percentage of contact meets manufacturer's recommendations.

   ° All connecting-rod bolts should be lubricated in accordance with the engine manufacturer's instructions and torqued to the specifications of the manufacturer. The lengths of the two pairs of bolts above the crankpin should be measured ultrasonicly pre- and post-tensioning.

   ° If connecting-rod bolt stretch was measured ultrasonicly during reassembly following the preservice inspection, the lengths of the two pairs of bolts above the connecting rod should be remeasured ultrasonicly before the link rod box is disassembled. Alternatively, the breakway torque should be measured. If bolt tension determined by either method is less than 93% of the value at installation, the cause should be determined, appropriate corrective action should be taken, and the interval between checks of bolt torque should be reevaluated.

   ° All connecting-rod bolts should be visually inspected for thread damage (e.g., galling), and the two pairs of connecting rod bolts above the crankpin should be inspected by magnetic particle testing (MT) to verify the continued absence of cracking. All washers used with the bolts should be examined visually for signs of galling or cracking, and replaced if damaged.

   ° A visual inspection should be performed of all external surfaces of the link rod box to verify the absence of any signs of service induced distress.

° All of the bolt holes in the link rod box should be inspected for thread damage (e.g., galling) or other signs of abnormalities. In addition, the bolt holes subject to the highest stresses (i.e., the pair immediately above the crankpin) should be examined with an appropriate nondestructive method to verify the continued absence of cracking. Any indications should be recorded for engineering evaluation and appropriate corrective action.

3. (a) Cylinder blocks shall be inspected at intervals calculated using the cumulative damage index (CDI) model and using inspection methodologies described by Failure Analysis Associates, Inc., (FaAA) in a report entitled "Design Review of TDI R-4 and RV-4 Series Emergency Diesel Generator Cylinder Blocks" (FaAA-84-9-11) dated December, 1984. In addition to these inspections, liquid penetrant inspection of the cylinder liner landing area should be performed anytime liners are removed. If inspection reveals cracks in the cylinder block between stud holes of adjacent cylinders, this condition shall be reported promptly to the NRC staff and the affected engine shall be considered inoperable. The engine shall not be restored to "operable" status until the proposed disposition and/or corrective actions have been approved by the NRC staff.

   (b) Prior to restart from the first refueling outage, Duke Power Company shall submit its cumulative damage analysis performed in accordance with FaAA report No. FaAA-84-9-11 dated December 1984, which verifies the acceptability of the "as-built" dimensions of the Catawba Unit 2 cylinder blocks. Alternatively, the block dimensions should be modified as necessary to meet the latest TDI specifications.

4. The engines shall be rolled over with the airstart system and the cylinder stopcocks open prior to any planned starts, unless that start occurs within 4 hours of a shutdown. The engines shall also be rolled over with the air-start system and the cylinder stopcocks open after 4 hours, but no more than 8 hours after engine shutdown and then rolled over once again approximately 24 hours after each shutdown. In the event an engine is removed from service for any reason other than the rolling over procedure prior to expiration of the 8 hour or 24 hour periods noted above, that engine need not be rolled over while it is out of service. Duke Power Company shall air roll the engine over with the stopcocks open at the time it is returned to service. The origin of any water detected in the cylinders must be determined and any cylinder head which leaks due to a crack shall be replaced. No cylinder heads that contain a through-wall weld repair where the repair was performed from one side only shall be used on the engines except for cylinder heads containing full penetration weld repairs as described in TDI drawing 102718, Revision 0.

5. Periodic inspections of the turbochargers shall include the following:

   ° The turbocharger thrust bearings should be visually inspected for excessive wear after 40 non-prelubed starts since the previous visual inspection.

   ° Turbocharger rotor axial clearance should be measured at each refueling outage to verify compliance with TDI/Elliott specifications. In addition, thrust bearing measurements should be compared with measurements taken previously to determine whether a trend exists. Any such trends shall be evaluated by Duke Power Company to determine need for further inspection or corrective action.

   ° Spectrographic and ferrographic engine oil analysis shall be performed quarterly to provide early evidence of bearing degradation. Particular attention should be paid to copper level and particulate size which could signify thrust bearing degradation.

   ° The nozzle ring components and inlet guide vanes should be visually inspected at each refueling outage for missing parts or parts showing distress. If such are noted, the entire ring assembly should be replaced.

   ° Pre-turbine exhaust temperature shall be monitored during engine operation to ensure that the manufacturer's temperature limit is not exceeded.

6. Main bearing No. 7 of emergency diesel generator 2B shall be disassembled and inspected at each refueling outage, both visually and with liquid penetrant, to verify that the bearings are free of distress. Subsequent to reassembly, run-in testing shall be performed in accordance with manufacturer's recommendations.

7. Operation beyond the first refueling outage shall require staff approval based on the staff's final review of the Owners Group generic findings and of the overall implementation status of Owners Group recommendations at Catawba Unit 2. This will include staff review of implementation status relative to open items identified in Sections 8.3.1.1.2(A) and 8.3.1.1.2(C) of SSER #5.

8. The following confirmatory information shall be submitted to the NRC staff prior to initial plant criticality.

   a. Verify that each engine base has been fabricated from normal class 40 gray iron which is free of Widmanstaetten graphite microstructure.

   b. Submit details concerning nature and cause of indication found on one rocker arm capscrew from Engine 2B. This information should address whether indication is service induced or whether it occurred as a result of fabrication or installation.

c. Submit evaluation of causal factors leading to wear of turbocharger thrust bearings in Engine 2B. Confirm that these causal factors have been found to be unique to the Engine 2B turbocharger and justify how this conclusion was reached.

d. Confirm that rotor float measurements have been conducted for both engine 2A turbochargers and that these measurements are acceptable per the TDI/Elliott specifications.

e. Verify implementation of TDI Service Information Memorandum (SIM) 300.

9. The number 7 main bearing from engine 2B shall be disassembled and inspected, both visually and with liquid penetrant, following a 100 hour endurance test of this bearing to verify that the bearing continues to be in adequate condition and free of any significant distress. The staff should be immediately notified of any adverse findings as a result of this inspection. A report shall be submitted to the NRC staff prior to initial plant criticality which documents in detail (1) the circumstances of the earlier failures of the No. 7 bearing, (2) the investigations, analyses, and inspections conducted to establish the cause of these failures, (3) the findings from these efforts, (4) the corrective actions taken, and (5) a description and the results of the 100 hour confirmatory test/inspection of the No. 7 bearing.

NUREG-1182

Technical Specifications

Catawba Nuclear Station, Unit Nos. 1 and 2

Docket Nos. 50-413 and 50-414

Appendix "A" to
License Nos. NPF-35 and NPF-48

**Issued by the
U.S. Nuclear Regulatory
Commission**

Office of Nuclear Reactor Regulation

February 1986

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG-1182

Technical Specifications

Catawba Nuclear Station, Unit Nos. 1 and 2

Docket Nos. 50-413 and 50-414

Appendix "A" to
License Nos. NPF-35 and NPF-48

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February 1986

BASES

SECTION PAGE

SECTION | PAGE

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

SECTION | PAGE

SECTION 1.0

DEFINITIONS

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

ACTION

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

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

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

AXIAL FLUX DIFFERENCE

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of 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 ALTERATION

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

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 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.

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

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

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.15 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

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

OFFSITE DOSE CALCULATION MANUAL

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

1.20 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

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

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

PURGE - PURGING

1.23 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

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

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR BUILDING INTEGRITY

1.26 REACTOR BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed,

b. The Annulus Ventilation System is OPERABLE, and

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

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 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.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length 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

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

SLAVE RELAY TEST

1.31 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.

SOLIDIFICATION

1.32 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.33 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

1.34 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

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

1.38 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.

VENTILATION EXHAUST TREATMENT SYSTEM

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

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.

WASTE GAS HOLDUP SYSTEM

1.41 A WASTE GAS HOLDUP 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.

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	≥ 0.99	$> 5\%$	$\geq 350°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3. HOT STANDBY	< 0.99	0	$\geq 350°F$
4. HOT SHUTDOWN	< 0.99	0	$350°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

*Excluding decay heat.

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

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 Figure 2.1-1 for four loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2:

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

MODES 3, 4, and 5:

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

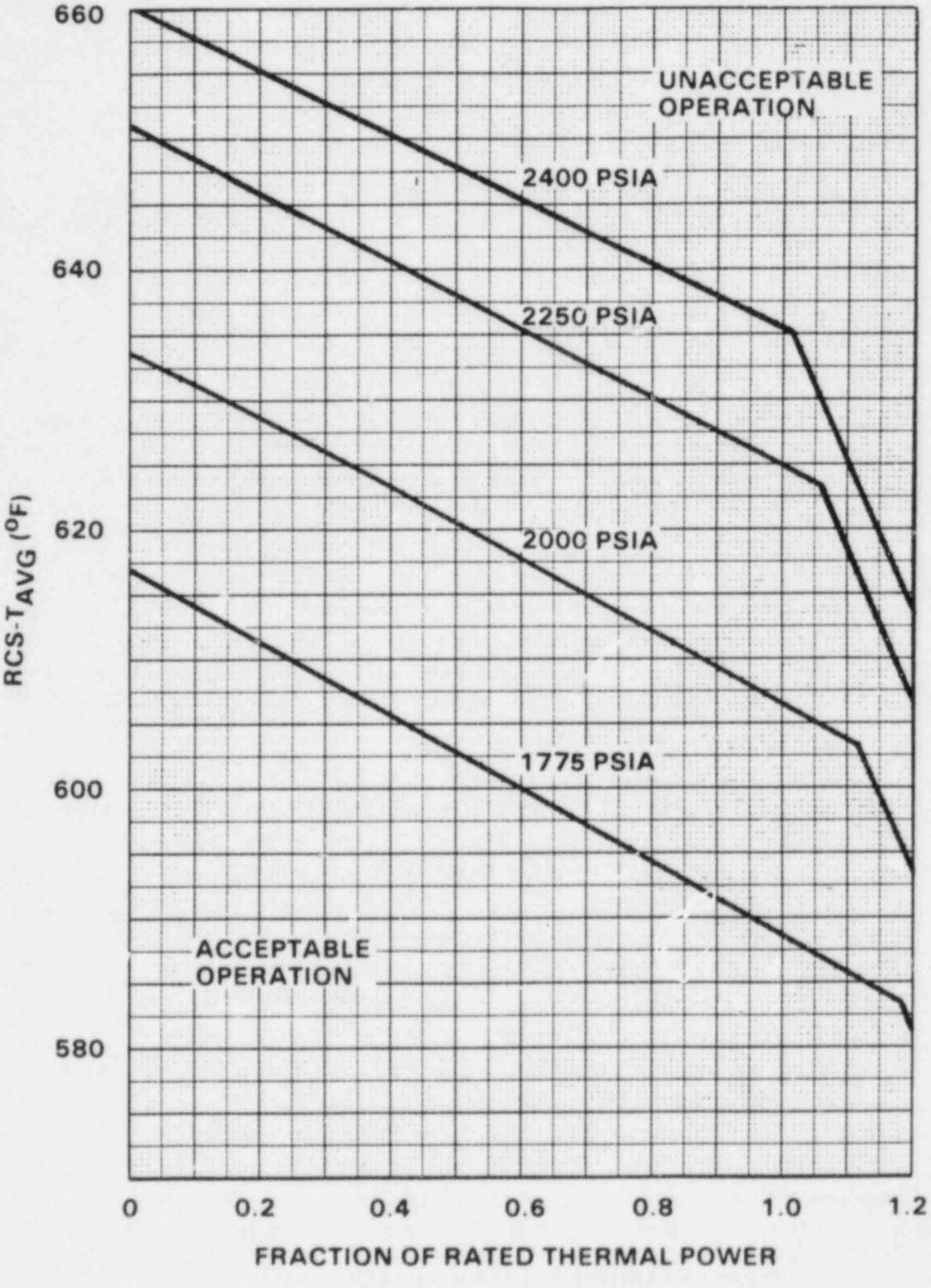

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlocks Setpoints shall be 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 Values column of Table 2.2-1, either:

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

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3 1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

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

Where:

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

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

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

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

TABLE 2.2.-1
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	≤10^5 cps	≤1.4×10^5 cps
7. Overtemperature ΔT	7.2	4.47	2.03	See Note 1	See Note 2
8. Overpower ΔT	4.3	1.3	1.2	See Note 3	See Note 4
9. Pressurizer Pressure-Low	4.0	2.21	1.5	≥1945 psig	≥1938 psig***
10. Pressurizer Pressure-High	7.5	4.96	0.5	≤2385 psig	≤2399 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.77	0.6	≥90% of loop design flow**	≥89.2% of loop design flow**

*RTP = RATED THERMAL POWER
**Loop design flow = 96,900 gpm
***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low					
a. Unit 1	17	14.2	1.5	≥17% of span from 0% to 30% RTP* increasing linearly to ≥ 54.9% of span from 30% to 100% RTP*	≥15.3% of span from 0% to 30% RTP* increasing linearly to ≥53.2% of span from 30% to 100% RTP*
b. Unit 2	17	14.2	1.5	≥17% of narrow range span	≥15.3% of narrow range span
14. Undervoltage - Reactor Coolant Pumps	8.57	0	1.0	≥77% of bus voltage (5082 volts) with a 0.7s response time	≥76% (5016 volts)
15. Underfrequency - Reactor Coolant Pumps	4.0	0	1.0	≥56.4 Hz with a 0.2s response time	≥55.9 Hz
16. Turbine Trip					
a.1 Control Valve EH Pressure - Low (Unit 1)	N.A.	N.A.	N.A.	≥550 psig	≥500 psig
a.2 Stop Valve EH Pressure - Low (Unit 2)	N.A.	N.A.	N.A.	≥550 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	(TA)	Z	TOTAL ALLOWANCE (S)	TRIP SETPOINT	SENSOR ERROR ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	$\leq$10% of RTP*	$\leq$12.2% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	$\leq$10% RTP* Turbine Impulse Pressure Equivalent	$\leq$12.2% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	$\leq$48% of RTP*	$\leq$50.2% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	$\leq$69% of RTP*	$\leq$70% of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	$\geq$10% of RTP*	$\geq$7.8% of RTP*
f. Power Range Neutron Flux, Not P-10	N.A.	N.A.	N.A.	$\leq$10% of RTP*	$\leq$12.2% of RTP*
g. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	$\leq$10% RTP* Turbine Impulse Pressure Equivalent	$\leq$12.2% RTP* Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.411;
K_2	=	0.02401/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	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}, $\tau_4 = 28$ s, $\tau_5 = 4$ s;
T	=	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, $\tau_6 = 0$;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T' $\leq$ 590.8°F (Nominal T_{avg} allowed by Safety Analysis)

K_3 = 0.001189;

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

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

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

(i) For $q_t - q_b$ between -43% and -6.5%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ is more negative than -43%, the ΔT Trip Setpoint shall be automatically reduced by 2% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ is more positive than -6.5%, the ΔT Trip Setpoint shall be automatically reduced by 1.641% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.4%.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

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 = As defined in Note 1,

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

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = 1.0704,

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

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

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, τ_7 = 10 s,

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

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.001707/°F for T > 590.8°F and K_6 = 0 for T $\leq$ 590.8°F,

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 590.8°F),

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.6%.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio, (DNBR), is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

This curve is based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49 \, [1 + 0.3 \, (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire Reactor Coolant System is hydrotested at 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the accident analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for all rod ejection accidents.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active or automatically blocked when P-10 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature-induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.2-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature-induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, is automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection and prevents DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER; with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 69% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trips.

P-9 On increasing power P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.

P-10 On increasing power P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

SECTIONS 3.0 AND 4.0

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

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 Condition for Operation refers to systems or components which are shared by both units, the ACTION requirements will apply to both units simultaneously. This will be indicated in the ACTION section;

b. Whenever the Limiting Condition 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 or footnotes. (For example, "...flow rate of 54,000 cfm (Unit 1) or 43,000 cfm (Unit 2)...").

SURVEILLANCE REQUIREMENTS

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

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any three consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

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

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

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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Δk/k for four loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Δk/k:

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

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

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

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3 or 4, 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% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.1.3 At least once per 18 months, the Reactor Makeup Water pumps shall be demonstrated OPERABLE by verifying a total combined flow rate of less than or equal to 240 gpm and a flow rate of less than or equal to 120 gpm for each pump.

4.1.1.1.4 At least once per 31 days, while in MODE 4, one Reactor Makeup Water pump shall be demonstrated inoperable by verifying that the motor circuit breaker is secured in the open position.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% $\Delta k/k$:

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 SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

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

   1) Reactor Coolant System boron concentration,

   2) Control rod position,

   3) Reactor Coolant System average temperature,

   4) Fuel burnup based on gross thermal energy generation,

   5) Xenon concentration, and

   6) Samarium concentration.

4.1.1.2.2 At least once per 18 months, each Reactor Makeup Water pump shall be demonstrated OPERABLE by verifying a flow rate of less than or equal to 120 gpm. At least once per 31 days, one Reactor Makeup Water pump shall be demonstrated inoperable by verifying that the motor circuit breaker is secured in the open position

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 $\Delta k/k/^{o}F$ for all the rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition; and

b. Less negative than -3.7×10^{-4} $\Delta k/k/^{o}F$ for all the rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 $\Delta k/k/^{o}F$ 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 prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, 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 of Specification 3.1.1.3a., above, 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 -2.8×10^{-4} $\Delta k/k/^{\circ}F$ (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 -2.8×10^{-4} $\Delta k/k/^{\circ}F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2#*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

#See Special Test Exceptions Specification 3.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the heated portion of the flow path is greater than or equal to 65°F when a flow path from the boric acid tanks is used, and

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the flow path from the boric acid tanks is greater than or equal to 65°F when it is a required water source;

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

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

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

*Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the Reactor Coolant System cold legs is less than or equal to 285°F.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying that a differential pressure across the pump of greater than or equal to 2380 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position, or that the discharge of each charging pump has been isolated from the Reactor Coolant System by at least two isolation valves with the power removed from the valve motor operators.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two* charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying that a differential pressure across each pump of greater than or equal to 2380 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one of more of the Reactor Coolant System cold legs is less than or equal to 285°F by verifying that the motor circuit breakers are secured in the open position or that the discharge of each charging pump has been isolated from the Reactor Coolant System by at least two isolation valves with power removed from the valve motor operators.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the Reactor Coolant System cold legs is less than or equal to 285°F.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 5100 gallons,

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1) A minimum contained borated water volume of 26,000 gallons,

2) A minimum boron concentration of 2000 ppm, and

3) A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

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

b. At least once per 24 hours by verifying the refueling water storage tank temperature when it is the source of borated water and the outside air temperature is less than 70°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System with:

   1) A minimum contained borated water volume of 19500 gallons,

   2) A minimum boron concentration of 7000 ppm, and

   3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

   1) A contained borated water volume of at least 363,513 gallons,

   2) A minimum boron concentration of 2000 ppm,

   3) A minimum solution temperature of 70°F, and

   4) A maximum solution temperature of 100°F.

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

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

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

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the refueling water storage tank temperature when the outside air temperature is either less than 70°F or greater than 100°F.

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

c. 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 of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuance 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.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

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

Single Rod Cluster Control Assembly Withdrawal at Full Power

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

Major Secondary Coolant System Pipe Rupture

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

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication 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 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.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicator agrees with the demand position indicator within 12 steps when exercised over the full-range of rod travel at least once per 18 months. The Reactor Trip System breakers can be closed in order to perform this surveillance.

*With the Reactor Trip System breakers in the closed position.

#See Special Test Exceptions Specification 3.10.5

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 3.3 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. 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.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

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

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

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

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 Specificat on 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 least once per 12 hours thereafter.

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

CONTROL BANK INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figure, or

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

SURVEILLANCE REQUIREMENTS

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

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

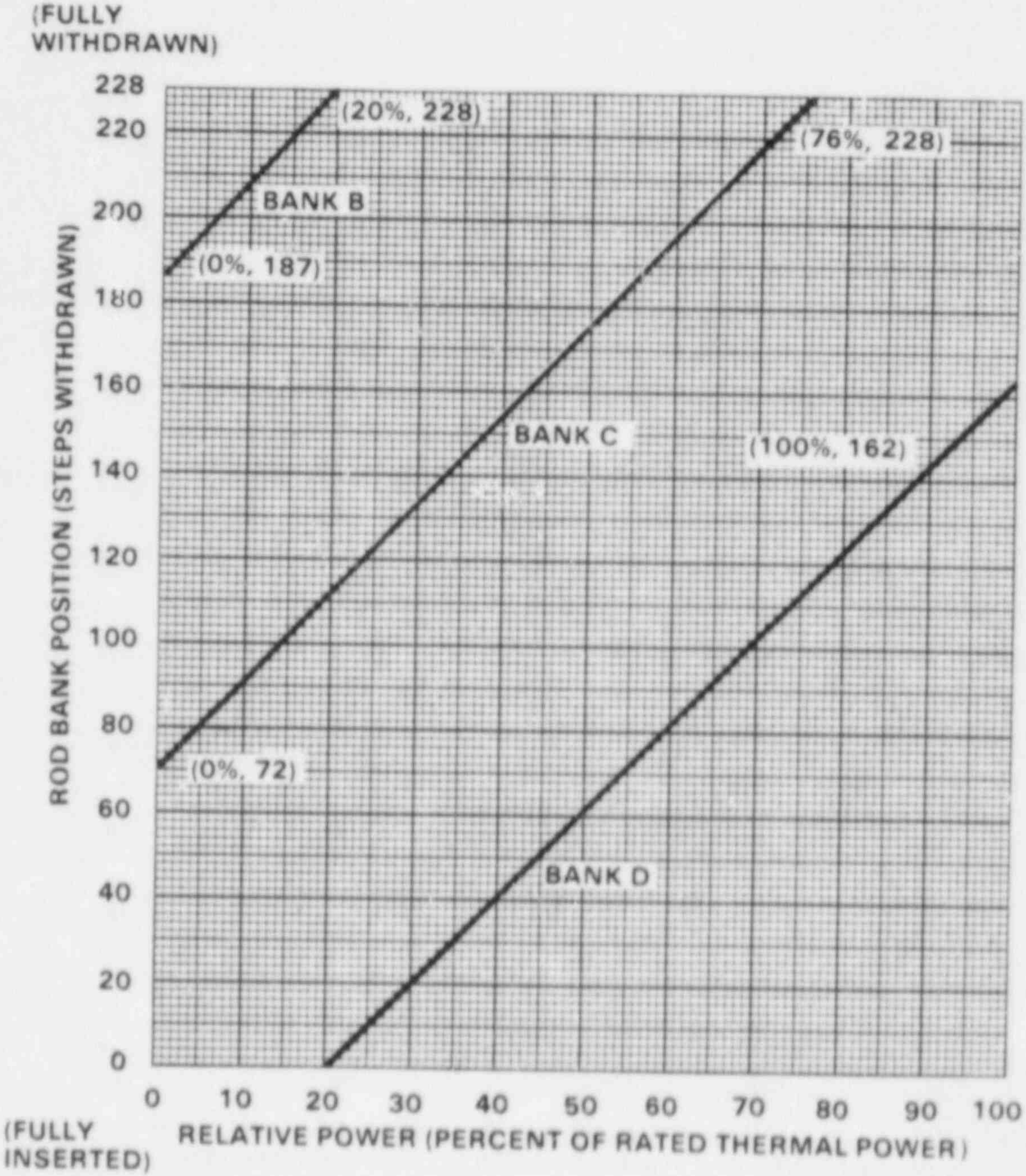

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

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 following target band (flux difference units) about the target flux difference:

a. ± 5% for Cycle 1 core average accumulated burnup of less than or equal to 5000 MWD/MTU;

b. + 3%, -9% for Cycle 1 core average accumulated burnup of greater than 5000 MWD/MTU; and

c. +3%, -12% for subsequent cycles.

The indicated AFD may deviate outside the above required target level at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

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

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation times during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

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

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*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. The Power Range Neutron Flux* - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

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:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

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

*Surveillance testing of the Power Range Neutron Flux Channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

SURVEILLANCE REQUIREMENTS (Continued)

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 pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

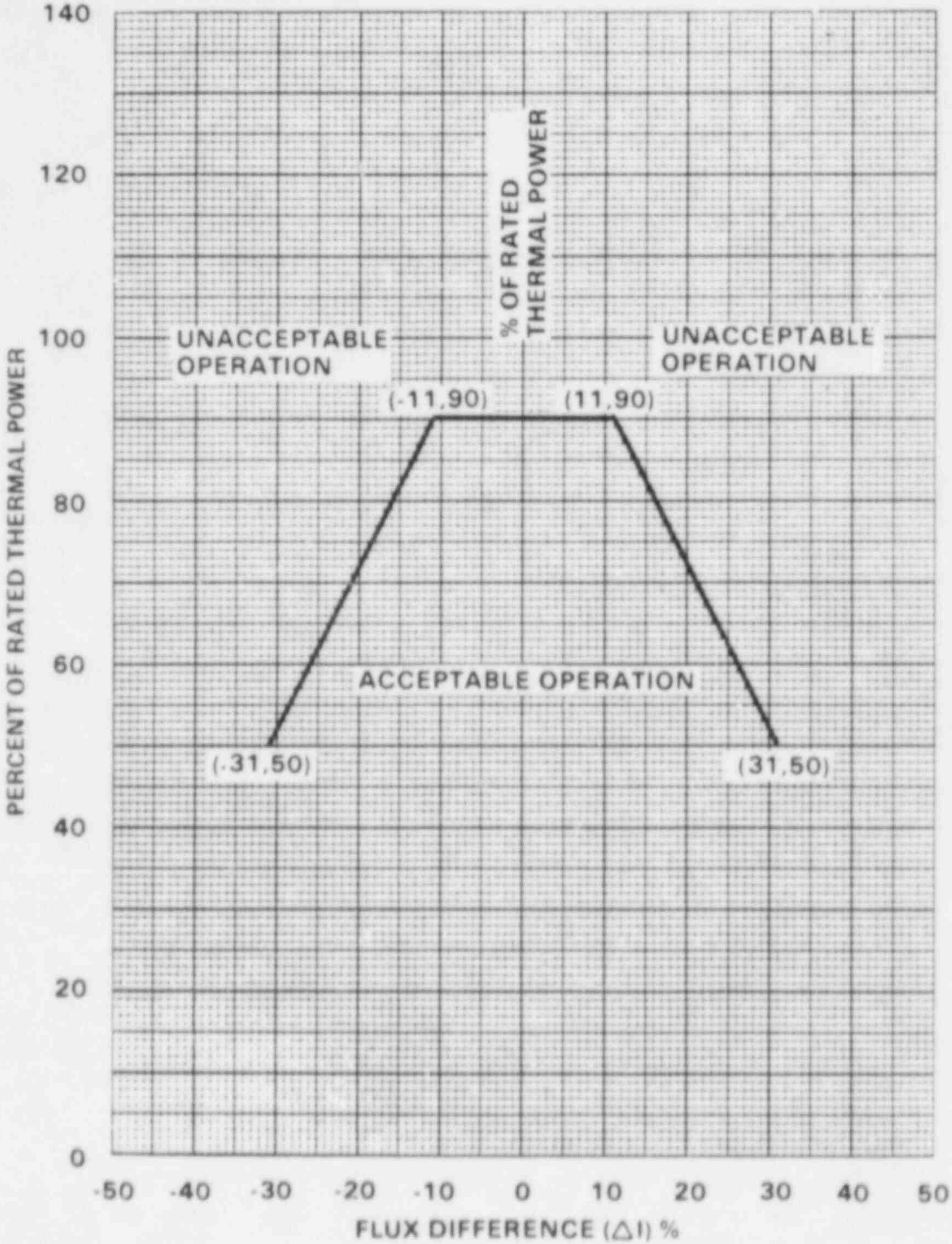

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5$$

Where: $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$K(Z)$ = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

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

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 incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

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:

$$F_{xy}^{L} = F_{xy}^{RTP}\ [1+0.2(1-P)],$$

Where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was 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 power distribution maps 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% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued)

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps 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 for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.9;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

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 power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

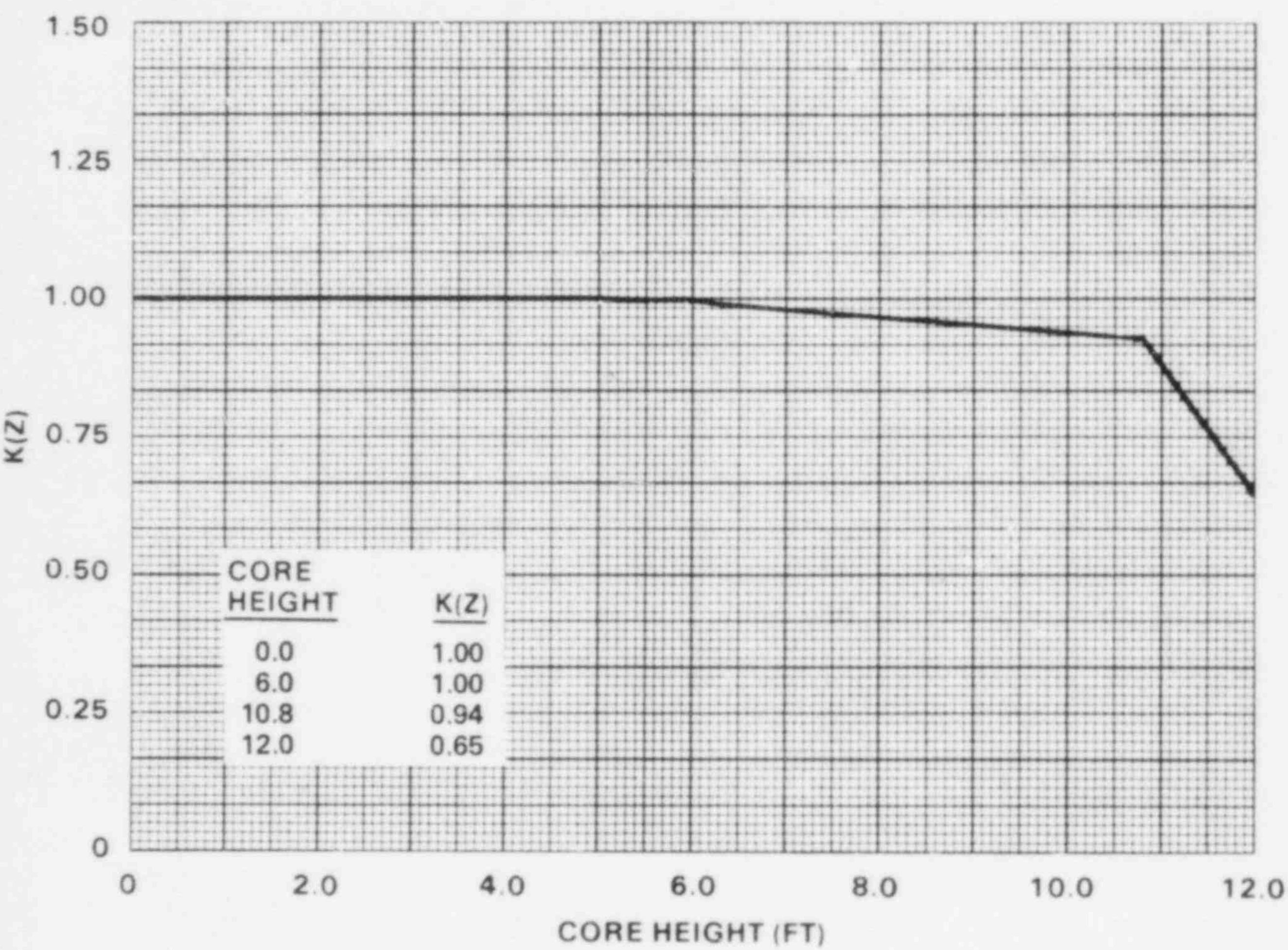

FIGURE 3.2-2

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

3/4.2.3 REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}$,

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

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 2.1% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of Reactor Coolant System total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

   1. Restore the combination of Reactor Coolant System total flow rate and R to within the above limits, or

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

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and Reactor Coolant System total flow rate comparison that the combination of R and Reactor Coolant System total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated Reactor Coolant System total flow rate are demonstrated, through incore flux mapping and Reactor Coolant System total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

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

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated Reactor Coolant System total flow rate determined by process computer readings or digital voltmeter measurement and R shall be determined to be within the region of acceptable operation of Figure 3.2-3:

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

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

4.2.3.3 The indicated Reactor Coolant System total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The Reactor Coolant System total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.3.5 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

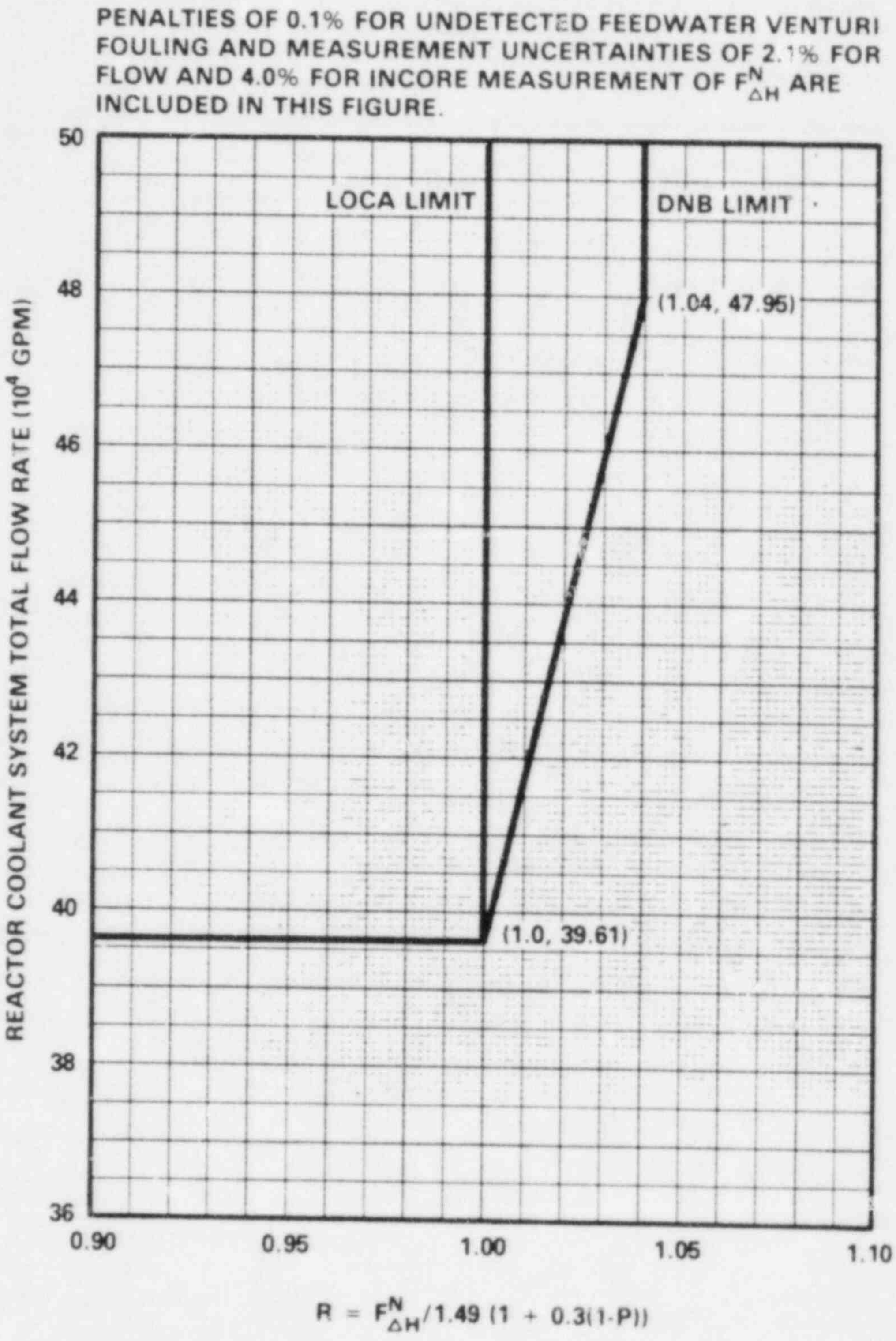

FIGURE 3.2-3
REACTOR COOLANT SYSTEM TOTAL FLOW RATE VERSUS R - FOUR LOOPS IN OPERATION

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 above 50% of RATED THERMAL POWER.

APPLICABILITY: MODE 1.*

ACTION:

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

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

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

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

   2. Within 2 hours either:

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

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

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

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

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*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

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

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

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

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS Four Loops in Operation
Indicated Reactor Coolant System T_{avg}	$\leq$ 592.5°F
Indicated Pressurizer Pressure	$\geq$ 2220 psig*

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

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	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. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
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	5
7. Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6#
8. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low	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
10. Pressurizer Pressure-High	4	2	3	1, 2	6#
11. Pressurizer Water Level-High	3	2	2	1	7#
12. Reactor Coolant Flow-Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#
13. Steam Generator Water Level--Low-Low	4/stm gen	2/stm gen in any operating stm gen	3/stm gen each operating stm gen	1, 2	6#
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6#
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6#
16. Turbine Trip					
a.1 Low Control Valve EH Pressure - (Unit 1)	4	2	3	1####	7#
a.2 Stop Valve EH Pressure - Low (Unit 2)	4	2	3	1####	7#
b. Turbine Stop Valve Closure	4	4	1	1####	11#
17. Safety Injection Input from ESF	2	1	2	1, 2	9

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. 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	8
f. Power Range Neutron Flux, Not P-10	4	3	4	1, 2	8
g. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
19. Reactor Trip Breakers	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

#The provisions of Specification 3.0.4 are not applicable.
##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
####Above the P-9 (Reactor Trip on Turbine Trip 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 at least HOT STANDBY within the next 6 hours.

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

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

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

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

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; or

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.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers, suspend all operations involving positive reactivity changes and verify Valves NV-231, NV-237, NV-241, and NV-244 are closed and secured in position within the next hour.

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

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

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

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive status light(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, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

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

ACTION 11 -With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	$\leq$ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	$\leq$ 4 seconds*
8. Overpower ΔT	$\leq$ 4 seconds
9. Pressurizer Pressure-Low	$\leq$ 2 seconds
10. Pressurizer Pressure-High	$\leq$ 2 seconds
11. Pressurizer Water Level-High	N.A.

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

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	$\leq$ 1 second
b. Two Loops (Above P-7 and below P-8)	$\leq$ 1 second
13. Steam Generator Water Level-Low-Low	$\leq$ 2.0 seconds
14. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 second
16. Turbine Trip	
a.1 Control Valve EH Pressure - Low (Unit 1)	N.A.
a.2 Stop Valve EH Pressure - Low (Unit 2)	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input from ESF	N.A.
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Automatic Trip and Interlock Logic	N.A.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip System Interlocks (Continued)						
e. Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1
f. Power Range Neutron Flux, Not P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1, 2
g. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M(8)	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

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 7 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 for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(5) Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

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

(8) With power greater than or equal to the interlock setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive status light.

(9) Monthly 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 status light. Monthly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to one-half decade (square root of 10) above background.

(10) Setpoint verification is not applicable.

(11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

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 Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3

ACTION:

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

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

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

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3.3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

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

Where:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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, Phase "A" Isolation, Feedwater Isolation, Control Room Area Ventilation Operation, Auxiliary Feedwater-Motor-Driven Pump, Purge & Exhaust Isolation, Annulus Ventilation Operation, Auxiliary Building Filtered Exhaust Operation, Emergency Diesel Generator Operation, Component Cooling Water, Turbine Trip, and Nuclear Service Water Operation)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure-Low	4	2	3	1, 2, 3#	19*
e. Steam Line Pressure-Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-High	4	2	3	1, 2, 3	16
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation (Nuclear Service Water Operation)					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure-High-High	4	2	3	1, 2, 3	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. Containment Isolation (Continued)					
c. Purge and Exhaust Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
4. Steam Line Isolation					
a. Manual Initiation					
1) System	2	1	2	1, 2, 3	22
2) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure-High-High	4	2	3	1, 2, 3	16
d. Steam Line Pressure-Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
e. Steam Line Pressure - Negative Rate-High	3/steam line	2/steam line in any steam line	2/steam line	3##	15*
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	27
b. Steam Generator Water Level-High-High (P-14)	4/steam generator	2/steam generator in any operating steam generator	3/steam generator in each operating steam generator	1, 2	19*
c. T_{avg}-Low (P-4 Interlock)	4	2	3	1, 2	19*
d. Doghouse Water Level-High	2	1	2	1, 2	27
e. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
6. Turbine Trip					
a. Manual Initiation	1	1	1	1,2	25
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1,2	27

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. Turbine Trip (Continued)					
c. Steam Generator Water Level-High-High (P-14)	4/steam generator	2/steam generator in any operating steam generator	3/steam generator in each operating steam generator	1,2	19*
d. Trip of All Main Feedwater Pumps	2/pump	1/pump	1/pump	1,2#	25
e. Reactor Trip (P-4)	2	2	2	1,2,3	22
f. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
7. Containment Pressure Control System					
a. Start Permissive	4/train	2/train	3/train	1, 2, 3, 4	19*
b. Termination	4/train	2/train	3/train	1, 2, 3, 4	19*
8. Auxiliary Feedwater					
a. Manual Initiation	1/train	1/train	1/train	1, 2, 3	26
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Auxiliary Feedwater (Continued)					
c. Stm. Gen. Water Level-Low-Low					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19*
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any two operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19*
d. Safety Injection-Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
e. Loss-of-Offsite Power-Start Motor-Driven Pumps and Turbine-Driven Pump	6-3/bus	2/bus either bus	2/bus	1, 2, 3	15*
f. Trip of All Main Feedwater Pumps-Start Motor-Driven Pumps	2/pump	1/pump	1/pump	1, 2#	25

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. Auxiliary Feedwater (Continued)					
g. Auxiliary Feedwater Suction Pressure-Low					
1) CAPS 5220, 5221, 5222	3/pump	2/pump	2/pump	1, 2, 3	15
2) CAPS 5230, 5231, 5232	3/pump	2/pump	2/pump	1, 2, 3	15
9. Containment Sump Recirculation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Refueling Water Storage Tank Level-Low	4	2	3	1, 2, 3, 4	16
Coincident With Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
10. Loss of Power					
a. 4 kV Bus Undervoltage-Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15*
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15*
11. Control Room Area Ventilation Operation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	All	24

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
11. Control Room Area Ventilation Operation (Continued)					
b. Loss-of-Offsite Power	3	2	2	1, 2, 3	15*
c. Safety Injection	See Item 1. above for all Safety Injections initiating functions and requirements.				
12. Containment Air Return and Hydrogen Skimmer Operation					
a. Manual Initiation	2	1	2	1,2,3,4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1,2,3,4	14
c. Containment Pressure-High-High	4	2	3	1,2,3	16
13. Annulus Ventilation Operation					
a. Manual Initiation	2	1	2	1,2,3,4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1,2,3,4	14
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
14. Nuclear Service Water Operation					
a. Manual Initiation	2	1	2	1,2,3,4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1,2,3,4	21

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
14. Nuclear Service Water Operation (Continued)					
c. Loss-of-Offsite Power	3	2	2	1,2,3	15*
d. Containment Spray	See Item 2. above for all Containment Spray initiating functions and requirements.				
e. Phase "B" Isolation	See Item 3.b. above for all Phase "B" Isolation initiating functions and requirements.				
f. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
g. Suction Transfer-Low Pit Level	2	1	2	1,2,3,4	21
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation)					
a. Manual Initiation	2	1	2	1,2,3,4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1,2,3,4	21
c. Loss-of-Offsite Power	3	2	2	1,2,3,4	15*
d. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
16. Auxiliary Building Filtered Exhaust Operation					
a. Manual Initiation	2	1	2	1,2,3,4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1,2,3,4	21

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Auxiliary Building Filtered Ventilation Exhaust Operation (Continued)					
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
17. Diesel Building Ventilation Operation					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	21
c. Emergency Diesel Generator Operation	See Item 15. above for all Emergency Diesel Generator Operation initiating functions and requirements.				
18. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Pressurizer Pressure, not P-11	3	2	2	1, 2, 3	20
c. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
d. Reactor Trip, P-4	2	2	2	1, 2, 3	22
e. Steam Generator Water Level, 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

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

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

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

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

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

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

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

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

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

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 20 - With less than the Minimum Channels OPERABLE, within 1 hour determine by observation of the associated permissive status light(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.4.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, restore the inoperable channel to OPERABLE status within 48 hours, or initiate and maintain operation of the Control Room Area Ventilation System with flow through the HEPA filters and carbon adsorbers.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 26 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 27 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Control Room Area Ventilation Operation, Auxiliary Feedwater-Motor-Driven Pump, Purge & Exhaust Isolation, Annulus Ventilation Operation, Auxiliary Building Filtered Exhaust Operation, Emergency Diesel Generator Operation, Component Cooling Water, Turbine Trip, and Nuclear Service Water Operation)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High	8.2	0.71	1.5	≤ 1.2 psig	≤ 1.4 psig
d. Pressurizer Pressure-Low	16.1	14.4	1.5	≥ 1845 psig	≥ 1839 psig
e. Steam Line Pressure-Low	4.6	1.31	1.5	≥ 725 psig	≥ 694 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	≤ 3 psig	≤ 3.2 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
b. Phase "B" Isolation (Nuclear Service Water Operation)					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure-High-High	12.7	0.71	1.5	$\leq$ 3 psig	$\leq$ 3.2 psig
c. Purge and Exhaust Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	≤ 3 psig	≤ 3.2 psig
d. Steam Line Pressure - Low	4.6	1.31	1.5	≥ 725 psig	≥ 694 psig*
e. Steam Line Pressure-Negative Rate - High	8.0	0.5	0	≤ 100 psi	≤ 122.8 psi**
5. Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)					
1. Unit 1	5.4	2.18	1.5	≤ 82.4% of narrow range instrument span	≤ 84.2% of narrow range instrument span
2. Unit 2	9.7	2.18	1.5	≤78.1% of narrow range instrument span	≤79.9% of narrow range instrument span
c. T_{avg}-Low	4.0	1.12	1.2	≥ 564°F	≥ 562°F
d. Doghouse Water Level-High	1.0	0	0.5	11 inches above 577' floor level	12 inches above 577' floor level
e. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Turbine Trip					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level-High-High (P-14)					
1. Unit 1	5.4	2.18	1.5	≤ 82.4% of narrow range instrument span	≤ 84.2% of narrow range instrument span
2. Unit 2	9.7	2.18	1.5	≤78.1% of narrow range instrument span	≤79.9% of narrow range instrument span
d. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
e. Reactor Trip (P-4)	N.A.	N.A.	N.A.	N.A.	N.A.
f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
7. Containment Pressure Control System					
a. Start Permissive	N.A.	N.A.	N.A.	≤ 0.4 psid	≤ 0.45 psid
b. Termination	N.A.	N.A.	N.A.	≥ 0.3 psid	≥ 0.25 psid
8. Auxiliary Feedwater					
a. Manual Initiation	N.A	N.A.	N.A.	N.A	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)					
c. Steam Generator Water Level - Low-Low					
1. Unit 1	17	14.2	1.5	≥ 17% of span from 0% to 30% RTP increasing linearly to ≥ 54.9% of span from 30% to 100% RTP	≥ 15.3% of span from 0% to 30% RTP increasing linearly to ≥ 53.2% of span from 30% to 100% RTP
2. Unit 2	17	14.2	1.5	≥17% of narrow range instrument span	≥15.3% of narrow range instrument span
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
e. Loss-of-Offsite Power	N.A.	N.A.	N.A.	≥ 3500 V	≥ 3200 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low					
1) CAPS 5220, 5221, 5222	N.A.	N.A.	N.A.	≥ 10.5 psig	≥ 9.5 psig
2) CAPS 5230, 5231, 5232					
a. Unit 1	N.A	N.A.	N.A.	≥ 6.2 psig	≥ 5.2 psig
b. Unit 2	N.A.	N.A.	N.A.	≥ 6.0 psig	≥ 5.0 psig
9. Containment Sump Recirculation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low	N.A.	N.A.	N.A.	≥ 177.15 inches	≥ 162.4 inches
Coincident With Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
10. Loss of Power					
a. 4 kV Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	≥ 3500 V	≥ 3200 V
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	N.A.	N.A.	N.A.	≥ 3685 V	≥ 3611 V
11. Control Room Area Ventilation Operation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Loss-of-Offsite Power	N.A.	N.A.	N.A.	≥ 3500 V	≥ 3200 V
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
12. Containment Air Return and Hydrogen Skimmer Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	≤ 3 psig	≤ 3.2 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Annulus Ventilation Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
14. Nuclear Service Water Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Loss-of-Offsite Power	N.A.	N.A.	N.A.	$\geq$ 3500 V	$\geq$ 3200 V
d. Containment Spray	See Item 2. above for all Containment Spray Setpoints and Allowable Values.				
e. Phase "B" Isolation	See Item 3.b. above for all Phase "B" Isolation Setpoints and Allowable Values.				
f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
g. Suction Transfer-Low Pit Level	NA	NA	NA	$\geq$El. 554.4 ft.	$\geq$El. 552.9 ft.
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation)					
a. Manual Initiation	N.A	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation) (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Loss-of-Offsite Power	N.A.	N.A.	N.A.	$\geq$ 3500 V	$\geq$ 3200 V
d. Safety Injection	See Item. 1 above for all Safety Injection Setpoints and Allowable Values.				
16. Auxiliary Building Filtered Exhaust Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
17. Diesel Building Ventilation Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Emergency Diesel Generator Operation	See Item 15. above for all Emergency Diesel Generator Operation Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	1955 psig	$\geq$1944 psig
b. Pressurizer Pressure, not P-11	N.A.	N.A.	N.A.	1955 psig	$\leq$1966 psig
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$553°F	$\geq$551°F
d. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
e. Steam Generator Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

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

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Purge and Exhaust Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Diesel Building Ventilation Operation	N.A.
h. Nuclear Service Water Operation	N.A.
i. Turbine Trip	N.A.
j. Component Cooling Water	N.A.
k. Annulus Ventilation Operation	N.A.
l. Auxiliary Building Filtered Exhaust Operation	N.A.
m. Reactor Trip	N.A.
n. Emergency Diesel Generator Operation	N.A.
o. Containment Air Return and Hydrogen Skimmer Operation	N.A.
p. Auxiliary Feedwater	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation[(2)]	$\leq 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	≤ 6
5) Auxiliary Feedwater[(5)]	N.A.
6) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	≤ 11
10) Control Room Area Ventilation Operation	N.A.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Containment Pressure-High (Continued)	
11) Annulus Ventilation Operation	≤ 23
12) Auxiliary Building Filtered Exhaust Operation	N.A.
13) Containment Sump Recirculation	N.A.
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation[(2)]	$\leq 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	≤ 6
5) Auxiliary Feedwater[(5)]	N.A.
6) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	≤ 11
10) Control Room Area Ventilation Operation	N.A.
11) Annulus Ventilation Operation	≤ 23
12) Auxiliary Building Filtered Exhaust Operation	N.A.
13) Containment Sump Recirculation	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12^{(3)}/22^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation[(2)]	$\leq 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	≤ 6
5) Auxiliary Feedwater[(5)]	≤ 60
6) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	≤ 11

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure-Low (Continued)	
10) Control Room Area Ventilation Operation	N.A.
11) Annulus Ventilation Operation	$\leq$ 23
12) Auxiliary Building Filtered Exhaust Isolation	N.A.
13) Containment Sump Recirculation	N.A.
b. Steam Line Isolation	$\leq$ 7
5. Containment Pressure-High-High	
a. Containment Spray	$\leq$ 45
b. Phase "B" Isolation	$\leq 65^{(3)}/76^{(4)}$
Nuclear Service Water Operation	N.A.
c. Steam Line Isolation	$\leq$ 7
d. Containment Air Return and Hydrogen Skimmer Operation	$\leq$ 600
6. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	$\leq$ 7
7. Steam Generator Water Level-High-High	
a. Turbine Trip	$\leq$ 3
b. Feedwater Isolation	$\leq$ 7
8. T_{avg}-Low	
Feedwater Isolation	N.A.
9. Doghouse Water Level-High	
Feedwater Isolation	N.A.
10. Start Permissive	
Containment Pressure Control System	N.A.
11. Termination	
Containment Pressure Control System	N.A.

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Steam Generator Water Level-Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
13. Loss-of-Offsite Power	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pumps	$\leq$ 60
c. Control Room Area Ventilation Operation	N.A.
d. Emergency Diesel Generator Operation	$\leq$ 11
1) Diesel Building Ventilation Operation	N.A.
2) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
14. Trip of All Main Feedwater Pumps	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine Trip	N.A.
15. Auxiliary Feedwater Suction Pressure-Low	
Auxiliary Feedwater (Suction Supply Automatic Realignment)	$\leq 15^{(6)}$
16. Refueling Water Storage Tank Level-Low	
Coincident with Safety Injection Signal (Automatic Switchover to Containment Sump)	$\leq$ 60
17. Loss of Power	
a. 4 kV Bus Undervoltage - Loss of Voltage	$\leq$ 8.5
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	$\leq$ 600
18. Suction Transfer-Low Pit Level	
Nuclear Service Water Operation	N.A.

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps, Safety Injection and residual heat removal pumps.

(2) Valves KC305B and KC315B are exceptions to the response times listed in the table. The following response times in seconds are the required values for these valves for the initiating signal and function indicated:

2.d	$\leq 30^{(3)}/40^{(4)}$
3.d	$\leq 30^{(3)}$
4.d	$\leq 30^{(3)}/40^{(4)}$

(3) Diesel generator starting and sequence loading delays <u>not</u> included. Offsite power available. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps.

(4) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish Safety Injection path and attainment of discharge pressure for centrifugal charging pumps.

(5) Response time for motor-driven auxiliary feedwater pumps on all Safety Injection signals shall be less than or equal to 60 seconds. Response time limit includes attainment of discharge pressure for auxiliary feedwater pumps.

(6) Response time includes a 5-second delay.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Control Room Area Ventilation Operation, Auxiliary Feedwater-Motor-Driven Pump, Purge and Exhaust Isolation, Annulus Ventilation Operation, Auxiliary Building Filtered Exhaust Operation, Emergency Diesel Generators Operation, Component Cooling Water, Turbine Trip, and Nuclear Service Water Operation)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
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.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3
5. Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Feedwater Isolation (Continued)								
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. T_{avg}-Low (P-4 Interlock)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
d. Doghouse Water Level-High	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
6. Turbine Trip								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
c. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
d. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
e. Reactor Trip (P-4)	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
f. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
7. Containment Pressure Control System								
a. Start Permissive	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Termination	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. 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 and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A	1, 2, 3
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
g. Auxiliary Feedwater Suction Pressure-Low	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
9. Containment Sump Recirculation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. Refueling Water Storage Tank Level - Low Coincident With	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Loss of Power								
a. 4 kV Bus Undervoltage-Loss of Voltage	N.A.	R	N.A.	M (2)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	N.A.	R	N.A.	M (2)	N.A.	N.A.	N.A.	1, 2, 3, 4
11. Control Room Area Ventilation Operation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	All
b. Loss-of-Offsite Power	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
12. Containment Air Return and Hydrogen Skimmer Operation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
13. Annulus Ventilation Operation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Annulus Ventilation Operation (Continued)								
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
14. Nuclear Service Water Operation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Loss-of-Offsite Power	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3
d. Containment Spray	See Item 2. above for all Containment Spray Surveillance Requirements.							
e. Phase "B" Isolation	See Item 3.b. above for all Phase "B" Isolation Surveillance Requirements.							
f. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
g. Suction Transfer-Low Pit Level	S	R	R	N.A.	N.A.	N.A.	N.A.	1, 2, 3
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation Nuclear Service Water Operation) (Continued)								
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Loss-of-Offsite Power	N.A.	R	N.A.	M(2)	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
16. Auxiliary Building Filtered Exhaust Operation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. 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	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
17. Diesel Building Ventilation Operation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Emergency Diesel Generator Operation	See Item 15. above for all Emergency Diesel Generator Operation Surveillance Requirements.							
18. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Pressurizer Pressure, not P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Generator Water Level, P-14	S	R	M	N.A.	M(1)	M(1)	Q	1, 2, 3

TABLE NOTATIONS

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Monthly testing shall consist of voltage sensor relay testing excluding actuation of load shedding diesel start, and time - delay timers.
(3) Monthly testing shall consist of relay testing excluding final actuation of the pumps or valves.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere - High Gaseous Radioactivity (Low Range - EMF-39)	1	1	All	***	30
b. Reactor Coolant System Leakage Detection					
1) Particulate Radioactivity (Low Range - EMF-38)	N.A.	1	1, 2, 3, 4	N.A.	33
2) Gaseous Radioactivity (Low Range - EMF-39)	N.A.	1	1, 2, 3, 4	N.A.	33
2. Fuel Storage Pool Areas					
a. High Gaseous Radioactivity (Low Range - EMF-42)	1	1	**	$\leq 1.7 \times 10^{-4}$ μCi/ml	34
b. Criticality-Radiation Level (Fuel Bridge - Low Range - 1EMF-15, 2EMF-4)	1	1	*	$\leq$ 15 mR/h	32
3. Control Room					
Air Intake-Radiation Level - High Gaseous Radioactivity (Low Range - EMF-43 A & B)	1/intake	2 (1/intake)	All	$\leq 1.7 \times 10^{-4}$ μCi/ml	31
4. Auxiliary Building Ventilation High Gaseous Radioactivity (Low Range - EMF-41)	1	1	1, 2, 3, 4	$\leq 1.7 \times 10^{-4}$ μCi/ml	35
5. Component Cooling Water System (EMF-46 A&B)	1	1	All	$\leq 1 \times 10^{-3}$ μCi/ml	36

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* With fuel in the fuel storage pool areas.

** With irradiated fuel in the fuel storage pool areas.

*** Trip Setpoint concentration value (μCi/ml) is to be established such that the actual submersion dose rate would not exceed 2 mR/h in the containment building. The Setpoint value may be increased up to the equivalent limits of Specification 3.11.2.1 in accordance with the methodology and parameters in the ODCM during containment purge or vent provided the Setpoint value does not exceed twice the maximum concentration activity in the containment determined by the sample analysis performed prior to each release in accordance with Table 4.11-2.

ACTION STATEMENTS

ACTION 30 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 31 - With the number of operable channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the affected Control Room Ventilation System intake from outside air with recirculating flow through the HEPA filters and carbon adsorbers.

ACTION 32 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel building.

ACTION 33 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 34 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, operation may continue provided the Fuel Handling Ventilation Exhaust System is operating and discharging through the HEPA filters and carbon adsorbers. Otherwise, suspend all operations involving fuel movement in the fuel building.

ACTION 35 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, operation may continue provided the Auxiliary Building Filtered Exhaust System is operating and discharging through the HEPA filter and carbon adsorbers.

ACTION 36 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity (gross gamma) at a lower limit of detection of no more than 10^{-7} μCi/ml.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere - High Gaseous Radioactivity (Low Range - EMF-39)	S	R	M	All
b. Reactor Coolant System Leakage Detection (Low Range - EMF-38 and Low Range - EMF-39)	S	R	M	1, 2, 3, 4
2. Fuel Storage Pool Areas				
a. High Gaseous Radioactivity (Low Range - EMF-42)	S	R	M	**
b. Criticality-Radiation Level (Fuel Bridge - Low Range - 1EMF-15, 2EMF-4)	S	R	M	*
3. Control Room				
Air Intake Radiation Level - High Gaseous Radioactivity - (Low Range - EMF-43 A & B)	S	R	M	All
4. Auxiliary Building Ventilation				
High Gaseous Radioactivity (Low Range - EMF-41)	S	R	M	1, 2, 3, 4
5. Component Cooling Water System (EMF-46 A&B)	S	R	M	All

TABLE NOTATIONS

* With fuel in the fuel storage pool area.

** With irradiated fuel in the fuel storage pool areas.

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

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

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

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by irradiating each detector used and determining the acceptability of its voltage curve for:

a. Recalibration of the Excore Neutron Flux Detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$, and F_{xy}.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above accessible seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. Data retrieved from the triaxial time-history accelerograph shall include a post-event CHANNEL CALIBRATION obtained by actuation of the internal test and calibrate function immediately prior to removing data. CHANNEL CALIBRATION shall be performed immediately after insertion of the new recording media in the triaxial time-history accelerograph recorder. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. 1MIMT 5070 (Remote Sensor A) Containment Base Slab	-1 g to + 1 g	1
b. 1MIMT 5080 (Remote Sensor B) Containment Vessel Elev 619'5"	-1 g to + 1 g	1
c. 1MIMT 5090 (Starter Unit) Containment Base Slab	0.005 g to 0.05 g	1
2. Triaxial Peak Accelerographs		
a. 1MIMT 5010 - Containment Bldg. Elev 613'8 9/16"	0 g to + 2 g	1
b. 1MIMT 5020 - Containment Bldg. Elev 567'2½"	0 g to + 2 g	1
c. 1MIMT 5030 - Auxiliary Bldg. Elev 543'	0 g to + 2 g	1
3. Triaxial Seismic Switch		
1MIMT 5000 - Containment Base Slab	0.025 g to 0.25 g	1*
4. Triaxial Response-Spectrum Recorders		
a. 1MIMT 5040 - Containment Base Slab	0 to 34 g at 2 to 25 Hz	1*
b. 1MIMT 5050 - Containment Bldg. Elev 579'3½"	0 to 34 g at 2 to 25 Hz	1
c. 1MIMT 5060 - Auxiliary Bldg. Elev 577'	0 to 34 g at 2 to 25 Hz	1

*With reactor control room indication.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Time-History Accelerographs			
a. 1MIMT 5070 (Remote Sensor A) Containment Base Slab	M*	R	SA
b. 1MIMT 5080 (Remote Sensor B) Containment Vessel Elev 619'5"	M*	R	SA
c. 1MIMT 5090 (Starter Unit) Containment Base Slab	N.A.	R	SA
2. Triaxial Peak Accelerographs			
a. 1MIMT 5010 - Containment Bldg. Elev 613' 8 9/16"	N.A.	R	N.A.
b. 1MIMT 5020 - Containment Bldg. Elev 567' 2½"	N.A.	R	N.A.
c. 1MIMT 5030 - Auxiliary Bldg. Elev 543'	N.A.	R	N.A.
3. Triaxial Seismic Switch			
1MIMT 5000 - Containment Base Slab**	M	R	SA
4. Triaxial Response-Spectrum Recorders			
a. 1MIMT 5040 - Containment Base Slab**	M	R	SA
b. 1MIMT 5050 - Containment Bldg. Elev 579' 3½"	N.A.	R	N.A.
c. 1MIMT 5060 - Auxiliary Bldg. Elev 577'	N.A.	R	N.A.

*Except seismic trigger.

**With reactor control room indications.

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed		
a. Meteorological Tower	Nominal Elev. 661'10"	1
b. Meteorological Tower	Nominal Elev. 768'10"	1
2. Wind Direction		
a. Meteorological Tower	Nominal Elev. 661'10"	1
b. Meteorological Tower	Nominal Elev. 768'10"	1
3. Air Temperature - ΔT		
Meteorological Tower	Nominal Elev. 768'10"-661'10"	1

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 661' 10"	D	SA
b. Nominal Elev. 768' 10"	D	SA
2. Wind Direction		
a. Nominal Elev. 661' 10"	D	SA
b. Nominal Elev. 768' 10"	D	SA
3. Air Temperature - ΔT		
Nominal Elev. 768'10" - 661'10"	D	SA

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown monitoring instrumentation channels given in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	Reactor Trip Switchgear	1/trip breaker	1/trip breaker
2. Reactor Coolant Loop A&B Hot Leg Temperature	Auxiliary Shutdown Control Panel	2	1*
3. Reactor Coolant Loop A&B Cold Leg Temperature	Auxiliary Shutdown Control Panel	2	1*
4. Pressurizer Pressure	Auxiliary Shutdown Control Panel	2	1
5. Pressurizer Level	Auxiliary Shutdown Control Panel	2	1
6. Steam Generator Pressure	Auxiliary Feedwater Pump Motor Control Panels A&B	1/steam generator	1/steam generator
7. Steam Generator Level	Auxiliary Feedwater Pump Motor Control Panels A&B	1/steam generator	1/steam generator
8. Auxiliary Feedwater Flow Rate	Auxiliary Feedwater Pump Motor Control Panels A&B	1/steam generator	1/steam generator

*Channel required to be OPERABLE during operation from remote shutdown panel.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N.A.
2. Reactor Coolant Loop A&B Hot Leg Temperature	M	R
3. Reactor Coolant Loop A&B Cold Leg Temperature	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Level	M	R
8. Auxiliary Feedwater Flow Rate	M	R

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels except the unit vent-high-high range area monitor, the steam relief valve exhaust radiation monitor, the containment atmosphere-high range radiation monitor, and the reactor coolant radiation level less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE Channels for the unit vent-high-high range area monitor, or the steam relief valve exhaust radiation monitor, or the containment atmosphere-high range radiation monitor, or the reactor coolant radiation level less than required by the Minimum Channels OPERABLE requirements, initiate an alternate method of monitoring the appropriate parameter(s) within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
8. Refueling Water Storage Tank Water Level	2	1
9. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator
10. Reactor Coolant System Subcooling Margin Monitor	1	1
11. PORV Position Indicator*	2/Valve	1/Valve
12. PORV Block Valve Position Indicator**	2/Valve	1/Valve
13. Pressurizer Safety Valve Position Indicator	1/Valve	1/Valve
14. Containment Sump Water Level (Wide Range)	2	1

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
15. In Core Thermocouples	4/core quadrant	2/core quadrant
16. Unit Vent - High-High Range Area Monitor (EMF-54)	N.A.	1
17. Steam Relief Valve Exhaust Radiation Monitor (1EMF-26, 27, 28 or 29 and 2EMF-10, 11, 12 or 13)	N.A.	1
18. Containment Area - High Range Radiation Monitor (EMF-53 A or B)	N.A.	1
19. Reactor Vessel Water Level	2	1
20. Reactor Coolant Radiation Level (EMF-48)	N.A.	1

TABLE NOTATIONS

* Not applicable if the associated block valve is in the closed position.

** Not applicable if the associated block valve is in the closed position and power is removed.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Auxiliary Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
11. PORV Position Indicator	M	R
12. PORV Block Valve Position Indicator	M	R
13. Pressurizer Safety Valve Position Indicator	M	R
14. Containment Sump Water Level (Wide Range)	M	R

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT (Continued)	CHANNEL CHECK	CHANNEL CALIBRATION
15. In Core Thermocouples	M	R
16. Unit Vent - High-High Range Area Monitor (EMF-54)	M	R
17. Steam Relief Valve Exhaust Radiation Monitor (1EMF-26, 27, 28 and 29 and 2EMF-10, 11, 12 and 13)	M	R
18. Containment Area - High Range Radiation Monitor (EMF-53 A&B)	M	R*
19. Reactor Vessel Water Level	M	R
20. Reactor Coolant Radiation Level (EMF-48)	M	R

*CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/h and a one point calibration check of the detector below 10R/h with an installed or portable gamma source.

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent Chlorine Detection Systems, with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

a. With one Chlorine Detection System inoperable, restore the inoperable system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Area Ventilation System with flow through the HEPA filters and charcoal adsorbers.

b. With both Chlorine Detection Systems inoperable, within 1 hour initiate and maintain operation of the Control Room Area Ventilation System with flow through the HEPA filters and charcoal adsorbers.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each Chlorine Detection System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

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

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each of the above required smoke detection or flame detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.8.2 Each of the above required heat detection instruments shall be demonstrated OPERABLE as follows:

a. For nonrestorable spot-type detectors, at least two detectors out of every hundred, or fraction thereof, shall be removed every 5 years and functionally tested. For each failure that occurs on the detectors removed, two additional detectors shall be removed and tested; and

SURVEILLANCE REQUIREMENTS (Continued)

b. For restorable spot-type heat detectors, at least one detector on each signal initiating circuit shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Different detectors shall be selected for each test. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.8.3 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
1	R.H.R. Pump 1B	GG-53	El.522 + 0	1	0	1	A
2	R.H.R. Pump 1A	FF-53	El.522 + 0	1	0	1	A
3	Cont. Spray Pump 1B	GG-54	El.522 + 0	3	0	3	A
4	Cont. Spray Pump 1A	GG-55	El.522 + 0	2	0	2	A
5	R.H.R. Pump 2B	GG-61	El.522 + 0	1	0	1	A
6	R.H.R. Pump 2A	FF-61	El.522 + 0	1	0	1	A
7	Cont. Spray Pump 2B	GG-60	El.522 + 0	3	0	3	A
8	Cont. Spray Pump 2A	GG-59	El.522 + 0	2	0	2	A
9	Aux. F. W. Pumps	BB-51	El.543 + 0	14	0	12(6)	A(B)
10	Mech. Pene. Room	JJ-52	El.543 + 0	3	0	3	A
11	Corridor/Cables	NN-51	El.543 + 0	6	0	5	A
12	Recip. Chg. Pump	JJ-53	El.543 + 0	1	0	[illegible]	A
13	Safety Inj Pump 1B	HH-53	El.543 + 0	1	0	[illegible]	A
14	Safety Inj Pump 1A	GG-53	El.543 + 0	1	0	1	A
15	Cent. Chg. Pump 1B	JJ-54	El.543 + 0	2	0	2	A
16	Cent. Chg. Pump 1A	JJ-55	El.543 + 0	2	0	2	A
17	Aisles/Cables	KK-56	El.543 + 0	18	0	18	A
18	Aisles/Cables	EE-55	El.543 + 0	6	0	6	A
19	AFW Pumps (Unit 2)	BB-63	El.543 + 0	14	0	12(6)	A(B)
20	Mech. Pene. Room	JJ-62	El.543 + 0	3	0	3	A
21	Aisles/Cables	NN-61	El.543 + 0	6	0	6	A
22	Recip. Chg. Pump	JJ-60	El.543 + 0	1	0	1	A
23	Safety Inj. Pump 2B	HH-60	El.543 + 0	1	0	1	A
24	Safety Inj. Pump 2A	GG-60	El.543 + 0	1	0	1	A
25	Cent. Chg. Pump 2B	JJ-59	El.543 + 0	2	0	2	A
26	Cent. Chg. Pump 2A	JJ-58	El.543 + 0	2	0	2	A
27	Aisles/Cables	KK-59	El.543 + 0	20	0	20	A
28	Aisles/Cables	EE-58	El.543 + 0	6	0	6	A
29	SW Gear Equip. Room	AA-50	El.560 + 0	7	0	0	A
30	Elect. Pene. Room	CC-50	El.560 + 0	8	0	0	A
31	Corridor/Cables	EE-53	El.560 + 0	5	0	5	A
32	Corridor/Cables	KK-52	El.560 + 0	8	0	8	A
33	Corridor/Cables	NN-54	El.560 + 0	10	0	10	A

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
34	Aisles/Cables	JJ-56	El.560 + 0	14	0	14	A
35	Motor Control Centers	GG-56	El.560 + 0	2	0	2	A
36	Cable Tray Access	FF-56	El.568 + 0	2	0	2	A
37	Equip. Batteries	DD-55	El.554 + 0	5	0	4	A
38	Equip. Batteries	CC-55	El.554 + 0	5	0	4	A
39	Battery Room	CC-56	El.554 + 0	17	0	0	A
41	SW Gear Equip. Room	AA-64	El.560 + 0	7	0	0	A
42	Elect. Pene. Room	CC-65	El.560 + 0	8	0	0	A
43	Corrider/Cables	FF-61	El.560 + 0	5	0	5	A
44	Aisles/Cables	KK-63	El.560 + 0	8	0	8	A
45	Aisles/Cables	NN-60	El.560 + 0	13	0	13	A
46	Aisles/Cables	HH-59	El.560 + 0	13	0	13	A
47	Motor Control Center	GG-58	El.560 + 0	2	0	2	A
48	Cable Tray Access	FF-58	El.560 + 0	2	0	2	A
49	Equip. Batteries	DD-60	El.560 + 0	5	0	4	A
50	Equip. Batteries	CC-60	El.560 + 0	5	0	4	A
51	Battery Room	CC-59	El.560 + 0	17	0	0	A
53	SW Gear Equip. Room	AA-49	El.577 + 0	7	0	0	A
54	Aisles/Cables	CC-50	El.577 + 0	10	0	0	A
55	Aisles/Cables	NN-52	El.577 + 0	9	0	9	A
56	Aisles/Cables	PP-55	El.577 + 0	13	0	13	A
57	Aisles/Cables	LL-55	El.577 + 0	11	0	11	A
58	Aisles/Cables	HH-55	El.577 + 0	21	0	21	A
59	Motor Control Center	EE-54	El.577 + 0	2	0	2	A
60	Cable Room	CC-56	El.574 + 0	18	0	15	A
62	SW Gear Equip. Room	AA-64	El.577 + 0	7	0	0	A
63	Elect. Pene. Room	CC-64	El.577 + 0	10	0	0	A
64	Aisles/Cables	PP-62	El.577 + 0	9	0	9	A
65	Aisles/Cables	PP-59	El.577 + 0	16	0	16	A
66	Aisles/Cables	LL-59	El.577 + 0	11	0	11	A
67	Aisles/Cables	HH-59	El.577 + 0	21	0	21	A
68	Motor Control Center	FF-60	El.577 + 0	2	0	2	A

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
69	Cable Room	CC-59	El.577 + 0	18	0	15	A
71	Elect Pene. Room	CC-51	El.594 + 0	10	0	0	A
72	Control Room	CC-56	El.594 + 0	23	0	6	A
73	Vent. Equip. Room	FF-56	El.594 + 0	9	0	0	A
74	Aisles/Cables	LL-56	El.594 + 0	25	0	25	A
76	Aisles/Cables	PP-54	El.594 + 0	15	0	15	A
79	Elect. Pene. Room	BB-63	El.594 + 0	11	0	0	A
80	Control Room	BB-59	El.594 + 0	22	0	6	A
81	Ven. Equip. Room	FF-58	El.594 + 0	12	0	0	A
82	Aisles/Cables	KK-58	El.594 + 0	27	0	28	A
84	Aisles/Cables	NN-58	El.594 + 0	17	0	17	A
89	Fuel Pool Area #1	PP-50	El.605 + 10	19	7	19	A
90	Fuel Pool Area (Unit 2)	PP-64	El.605 + 10	19	7	19	A
128	UHI Bldg.	HH-44	El.550 + 0	2	3	2	A
129	Fuel Pool Purge Room (Unit 1)	NN-50	El.631 + 6	6	0	6	A
130	UHI Bldg. (Unit 2)	HH-71	El.594 + 0	2	3	2	A
131	Reactor Bldg.	0°-45°	Bel. El.565 + 3	4	0	0	A
132	Reactor Bldg.	45°-90°	Bel. El.565 + 3	3	0	0	A
133	Reactor Bldg.	90°-135°	Bel. El.565 + 3	4	0	0	A
134	Reactor Bldg.	135°-180	Bel. El.565 + 3	5	0	0	A
135	Reactor Bldg.	180°-225°	Bel. El.565 + 3	4	0	0	A
136	Reactor Bldg.	270°-315°	Bel. El.565 + 3	3	0	0	A
137	Reactor Bldg.	315°-0°	Bel. El.565 + 3	8	0	0	A
138	Reactor Bldg.	0°-45°	Bel. El.586 + 3	6	0	0	A
139	Reactor Bldg.	45°-90°	Bel. El.586 + 3	4	0	0	A
140	Reactor Bldg.	90°-135°	Bel. El.565 + 3	3	0	0	A
141	Reactor Bldg.	135°-180	Bel. El.586 + 3	8	0	0	A
142	Reactor Bldg.	180°-225°	Bel. El.586 + 3	5	0	0	A
143	Reactor Bldg	315°-0°	Bel. El.586 + 3	5	0	0	A
144	Reactor Bldg.	0°-45°	Bel. El.593 + 2½	14	0	0	A
145	Reactor Bldg.	45°-90°	Bel. El.593 + 2½	17	0	0	A
146	Reactor Bldg.	90-135°	Bel. El.593 + 2½	11	0	0	A

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
147	Reactor Bldg.	135°-180°	Bel. El.593 + 2½	10	0	0	A
148	Reactor Bldg.	180°-225°	Bel. El.593 + 2½	2	0	0	A
149	Reactor Bldg.	315°-0°	Bel. El.593 + 2½	7	0	0	A
150	Reactor Bldg. (Unit 2)	0°-45°	Bel. El.565 + 3	4	0	0	A
151	Reactor Bldg. (Unit 2)	45°-90°	Bel. El.565 + 3	3	0	0	A
152	Reactor Bldg. (Unit 2)	90°-135°	Bel. El.565 + 3	4	0	0	A
153	Reactor Bldg. (Unit 2)	135°-180°	Bel. El.565 + 3	5	0	0	A
154	Reactor Bldg. (Unit 2)	180°-225°	Bel. El.565 + 3	3	0	0	A
155	Reactor Bldg. (Unit 2)	270°-315°	Bel. El.565 + 3	4	0	0	A
156	Reactor Bldg. (Unit 2)	315°-0°	Bel. El.565 + 3	6	0	0	A
157	Reactor Bldg. (Unit 2)	0°-45°	Bel. El.586 + 6	6	0	0	A
158	Reactor Bldg. (Unit 2)	45°-90°	Bel. El.586 + 6	4	0	0	A
159	Reactor Bldg. (Unit 2)	90°-135°	Bel. El.586 + 6	3	0	0	A
160	Reactor Bldg. (Unit 2)	135°-180°	Bel. El.586 + 6	8	0	0	A
161	Reactor Bldg. (Unit 2)	180°-225°	Bel. El.586 + 6	5	0	0	A
162	Reactor Bldg. (Unit 2)	315°-0°	Bel. El.586 + 6	5	0	0	A
163	Reactor Bldg. (Unit 2)	0°-45°	Bel. El.593 + 2½	13	0	0	A
164	Reactor Bldg. (Unit 2)	45°-90°	Bel. El.593 + 2½	17	0	0	A
165	Reactor Bldg. (Unit 2)	90°-135°	Bel. El.593 + 2½	13	0	0	A
166	Reactor Bldg. (Unit 2)	135°-180°	Bel. El.593 + 2½	10	0	0	A
167	Reactor Bldg. (Unit 2)	180°-225°	Bel. El.593 + 2½	2	0	0	A
168	Reactor Bldg. (Unit 2)	315°-0°	Bel. El.593 + 2½	7	0	0	A
169	RCP-1A	Reactor Bldg.	El.593 + 2½	0	0	1	A
170	RCP-1B	Reactor Bldg.	El.593 + 2½	0	0	1	A
171	RCP-1C	Reactor Bldg.	El.593 + 2½	0	0	1	A
172	RCP-1D	Reactor Bldg.	El.593 + 2½	0	0	1	A
173	RCP-2A	45°	Bel. El.593 + 2½	0	0	1	A
174	RCP-2B	135°	Bel. El.593 + 2½	0	0	1	A
175	RCP-2C	225°	Bel. El.593 + 2½	0	0	1	A
176	RCP-2D	315°	Bel. El.593 + 2½	0	0	1	A
177	Filter Bed Unit 1B	Reactor Bldg.	Bel. El.565 + 3	2	0	2	A
178	Filter Bed Unit 1A	Reactor Bldg.	Bel. El.565 + 3	2	0	2	A

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
179	Filter Bed Unit 2A	Reactor Bldg.	Bel. El.565 + 3	2	0	2	A
180	Filter Bed Unit 2B	Reactor Bldg.	Bel. El.565 + 3	2	0	2	A
181a	Annulus		El.561 + 0	0	0	1	A
181b	Annulus		El.583 + 0	0	0	1	A
181c	Annulus		El.604 + 0	0	0	1	A
181d	Annulus		El.629 + 5	0	0	1	A
181e	Annulus		El.649 + 5	0	0	1	A
181f	Annulus		El.664 + 0	0	0	1	A
182a	Annulus		El.561 + 0	0	0	1	A
182b	Annulus		El.583 + 0	0	0	1	A
182c	Annulus		El.604 + 0	0	0	1	A
182d	Annulus		El.629 + 5	0	0	1	A
182e	Annulus		El.649 + 5	0	0	1	A
182f	Annulus		El.664 + 0	0	0	1	A
183	Fuel Pool Purge Room (Unit 2)	NN-64	El.631 + 6	6	0	6	A
212	Aisles/Cables	GG-57	El.522 + 0	2	0	2	A
213	Aux. Batt. Room	AA-55	El.554 + 0	4	0	4	A
214	Aux. Cont. Pwr. Batt.	AA-59	El.560 + 0	4	0	4	A
215	D/G Corridor	BB-45	El.556 + 0	3	0	3	A
216	D/G Corridor	AA-45	El.556 + 0	2	0	2	A
217	D/G Corridor	CC-71	El.560 + 0	3	0	3	A
218	D/G Corridor	BB-71	El.560 + 0	2	0	2	A
219	Mech. Pen. Room	HH-52	El.577 + 0	6	0	6	A
220	Mech. Pen. Room	JJ-62	El.577 + 0	6	0	6	A
222	Airlock Access	JJ-51	El.605 + 10	1	0	1	A
224	Airlock Access (Unit 2)	JJ-63	El.605 + 10	1	0	1	A
225	RN Pump Structure	West Section	El.600 + 0	8	0	8	A
226	RN Pump Structure	East Section	El.600 + 0	8	0	8	A
231	Reactor Bldg. (Unit 1)	260°-303°	Bel. El.668 + 10	10	0	0	A
232	Reactor Bldg. (Unit 2)	260°-303°	Bel. El.668 + 10	10	0	0	A
184	HVAC Duct for Rooms 331 and 332	FF-53, 543 + 0		1(Duct)	0	0	A

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION	MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
185	HVAC Duct for Rooms 203, 205 205A, 206A, 206B, 207 and 209A	MM-60, 543 + 0	1(Duct)	0	0	A
186	HVAC Duct for Rooms 301, 302, 305, and 307	NN-60, 560 + 0	1(Duct)	0	0	A
RF1A	Diesel Generator 1A	EE-41, 556 + 0	0	0	0(10)	A(B)
RF1B	Diesel Generator 1B	AA-41, 556 + 0	0	0	0(10)	A(B)
RF2A	Diesel Generator 2A	EE-72, 556 + 0	0	0	0(10)	A(B)
RF2B	Diesel Generator 2B	AA-72, 556 + 0	0	0	0(10)	A(B)

*The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate tests.

**Function A: Early warning fire detection and notification only.
Function B: Actuation of fire suppression system and early warning and notification.

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.9 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the Loose-Part Detection Systems shall be demonstrated OPERABLE by performance of:

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

b. An ANALOG CHANNEL OPERATIONAL TEST except for verification of Setpoint at least once per 31 days, and

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

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7 why this inoperability was not corrected within the time specified.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-8.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Waste Liquid Discharge Monitor (Low Range - EMF-49)	1 per station	40
b. Turbine Building Sump Monitor (Low Range - EMF-31)	1	42
c. Steam Generator Water Sample Monitor (Low Range - EMF-34)	1	43
2. Continuous Composite Samplers and Sampler Flow Monitor Conventional Waste Water Treatment Line	1 per station	42
3. Flow Rate Measurement Devices		
a. Waste Liquid Effluent Line	1 per station	41
b. Conventional Waste Water Treatment Line	1 per station	41
c. Low Pressure Service Water Minimum Flow Interlock	1 per station	41

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify:

1. The discharge line valving, and

2. The manual portion of the computer input for the release rate calculations performed on the computer, or the entire release rate calculations if such calculations are performed manually.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microCurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microCurie/gram DOSE EQUIVALENT I-131.

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, gaseous effluent releases via the atmospheric vent valves (off-normal mode) may continue provided grab samples of steam generator water are analyzed for radioactivity for up to 30 days at a lower limit of detection of no more than 10^{-7} microCurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microCurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microCurie/gram DOSE EQUIVALENT I-131.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Waste Liquid Discharge Monitor (Low Range - EMF-49)	D	P	R(2)	Q(1)
b. Turbine Building Sump Monitor (Low Range - EMF-31)	D	M	R(2)	Q(1)
c. Steam Generator Water Sample Monitor (Low Range - EMF-34)	D	M	R(2)	Q(1)
2. Continuous Composite Samplers and Sampler Flow Monitor				
Conventional Waste Water Treatment Line	D(3)	N.A.	R	N.A.
3. Flow Rate Measurement Devices				
a. Waste Liquid Effluent Line	D(3)	N.A.	R	N.A.
b. Conventional Waste Water Treatment Line	D(3)	N.A.	R	N.A.
c. Low Pressure Service Water Minimum Flow Interlock	D(3)	N.A.	R	Q

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exist:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (Alarm only), or

c. Instrument indicates a downscale failure (Alarm only).

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radic ctive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7 why this inoperability was not corrected within the time specified.

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

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Low Range - EMF-50)	1 per station	*	45
b. Effluent System Flow Rate Measuring Device	1 per station	*	46
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitors	1/train per station	**	51
b. Oxygen Monitors	2/train per station	**	49
3. Condenser Evacuation System Noble Gas Activity Monitor (Low Range - EMF-33)	1	1, 2, 3, 4	47
4. Vent System			
a. Noble Gas Activity Monitor (Low Range - EMF-36)	1	*	47
b. Iodine Sampler (EMF-37)	1	*	50
c. Particulate Sampler (EMF-35)	1	*	50
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. Containment Purge System			
Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Low Range - EMF-39)	1	***	48
6. Containment Air Release and Addition System			
Noble Gas Activity Monitor - Providing Alarm (Low Range - EMF-39)	1	*	45

TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times except when the isolation valve is closed and locked.

** During WASTE GAS HOLDUP SYSTEM operation.

*** At all times.

ACTION STATEMENTS

ACTION 45 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release either:

a. Vent system noble gas activity monitor providing alarm and automatic termination of release (Low Range - EMF-36) has at least one channel OPERABLE, or

b. At least two independent samples of the tank's contents are analyzed, and at least two technically qualified members of the facility staff independently verify:

1. The discharge valve lineup, and

2. The manual portion of the computer input for the release rate calculations performed on the computer, or the entire release rate calculations if such calculations are performed manually.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

TABLE 3.3-13 (Continued)

TABLE NOTATIONS

ACTION 49 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both channels inoperable, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

ACTION 50 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 51 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Low Range - EMF-50)	P	P	R(3)	Q(1)	*
b. Effluent System Flow Rate Measuring Device	P	N.A.	R	N.A.	*
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System					
a. Hydrogen Monitors	D	N.A.	Q(4)	M	**
b. Oxygen Monitors	D	N.A.	Q(5)	M	**
3. Condenser Evacuation System					
Noble Gas Activity Monitor (Low Range - EMF-33)	D	M	R(3)	N.A.	1, 2, 3, 4
4. Vent System					
a. Noble Gas Activity Monitor (Low Range - EMF-36)	D	M	R(3)	Q(2)	*
b. Iodine Sampler (EMF-37)	W	N.A.	N.A.	N.A.	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Vent System (Continued)					
c. Particulate Sampler (EMF-35)	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	N.A.	*
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*
5. Containment Purge System					
Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Low Range - EMF-39)	D	P	R(3)	Q(1)	***
6. Containment Air Release and Addition System					
Noble Gas Activity Monitor-Providing Alarm (Low Range - EMF-39)	D	P	R(3)	Q(1)	*

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times except when the isolation valve is closed and locked.
** During WASTE GAS HOLDUP SYSTEM operation.
*** At all times.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (Alarm only), or

c. Instrument indicates a downscale failure (Alarm only).

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annuciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples in accordance with the manufacturer's recommendations. In addition, a standard gas sample of nominal four volume percent hydrogen, balance nitrogen, shall be used in the calibration to check linearity of the hydrogen analyzer.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples in accordance with the manufacturer's recommendations. In addition, a standard gas sample of nominal four percent oxygen, balance nitrogen, shall be used in the calibration to check linearity of the oxygen analyzer.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam line inoperable and/or with one intermediate stop valve or one intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 7 days while in MODE 1 and while in MODE 2 with the turbine operating, by cycling each of the following valves through at least one complete cycle from the running position:

   1) Four high pressure turbine stop valves,

   2) Six low pressure turbine intermediate stop valves, and

   3) Six low pressure turbine intercept valves.

b. At least once per 31 days while in MODE 1 and while in MODE 2 with the turbine operating, by direct observation of the movement of each of the above valves and the four high pressure turbine control valves, through one complete cycle from the running position,

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

d. At least once per 40 months by disassembling at least one of each of the above valves (including the four high pressure turbine control valves) and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

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 least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE and at least two of these reactor coolant loops shall be in operation:*

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

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

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, 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, restore at least two loops to operation within 72 hours or open the Reactor Trip System breakers.

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 12% at least once per 12 hours.

4.4.1.2.3 At least two reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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

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. Residual Heat Removal Loop A, and

f. Residual Heat Removal Loop B.

APPLICABILITY: MODE 4.

ACTION:

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

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

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

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 285°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), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

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

4.4.1.3.3 At least one reactor coolant or residual heat removal loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

a. One additional residual heat removal loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 12%.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

One residual heat removal loop may be inoperable for up to 2 hours for surveillance testing provided the other residual heat removal 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 285°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.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one residual heat removal loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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

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

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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 1656 cubic feet and at least two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With more than one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply the ACTION b. or c. above, as appropriate, for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

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

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

4.4.4.3 (Unit 1 only) The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive and control power from the normal to the emergency power supply, and

b. Operating the valves through a complete cycle of full travel.

4.4.4.4 (Unit 2 only) The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive (N_2) and control power from the normal to the emergency power supply,

b. Isolating and venting the normal (air) supply, and

c. Operating the valves through a complete cycle of full travel.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

SURVEILLANCE REQUIREMENTS (Continued)

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Table 4.4-1
MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Four	Four
First Inservice Inspection	All	Two
Second & Subsequent Inservice Inspections	One[1]	One[2]

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N. A.	N. A.	N. A.	N. A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N. A.	N. A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N. A.	N. A.
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N. A.	N. A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N. A.	N. A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N. A.	N. A.

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous Radioactivity Monitoring System,

b. The Containment Floor and Equipment Sump Level and Flow Monitoring Subsystem, and

c. Either the Containment Ventilation Unit Condensate Drain Tank Level Monitoring Subsystem or the Containment Atmosphere Particulate Radioactivity Monitoring System.

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

ACTION:

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Floor and Equipment Sump Level and Flow Monitoring Subsystem-performance of CHANNEL CALIBRATION at least once per 18 months, and

c. Containment Ventilation Unit Condensate Drain Tank Level Monitoring Subsystem-performance of CHANNEL CALIBRATION at least once per 18 months.

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

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

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

f. 1 gpm leakage 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, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the 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.

*Until 0048 hours, October 23, 1985, operation in Modes 3 and 4 is permitted with the Reactor Coolant System unidentified leakage rate > 1 gpm but < 5 gpm. If the unidentified leakage rate is not reduced to < 1 gpm by the above time, the unit will be placed in COLD SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the containment atmosphere gaseous radioactivity monitor at least once per 12 hours;

b. Monitoring the containment floor and equipment sumps inventory and discharge at least once per 12 hours;

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

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

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

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

a. At least once per 18 months,

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
NI59	Accumulator Discharge
NI60	Accumulator Discharge
NI70	Accumulator Discharge
NI71	Accumulator Discharge
NI81	Accumualtor Discharge
NI82	Accumulator Discharge
NI93	Accumulator Discharge
NI94	Accumulator Discharge
NI124	Safety Injection (Hot Leg)
NI125	Residual Heat Removal (Hot Leg)
NI126	Safety Injection (Hot Leg)
NI128	Safety Injection (Hot Leg)
NI129	Residual Heat Removal (Hot Leg)
NI134	Safety Injection (Hot Leg)
NI156	Safety Injection (Hot Leg)
NI157	Safety Injection (Hot Leg)
NI159	Safety Injection (Hot Leg)
NI160	Safety Injection (Hot Leg)
NI165	Safety Injection/Residual Heat Removal (Cold Leg)
NI167	Safety Injection/Residual Heat Removal (Cold Leg)
NI169	Safety Injection/Residual Heat Removal (Cold Leg)
NI171	Safety Injection/Residual Heat Removal (Cold Leg)
NI175	Safety Injection/Residual Heat Removal (Cold Leg)
NI176	Safety Injection/Residual Heat Removal (Cold Leg)
NI180	Safety Injection/Residual Heat Removal (Cold Leg)
NI181	Safety Injection/Residual Heat Removal (Cold Leg)
NI248	Upper Head Injection
NI249	Upper Head Injection
NI250	Upper Head Injection
NI251	Upper Head Injection
NI252	Upper Head Injection
NI253	Upper Head Injection

TABLE 3.4-1 (Continued)

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
ND1B*#	Residual Heat Removal
ND2A*#	Residual Heat Removal
ND36B*#	Residual Heat Removal
ND37A*#	Residual Heat Removal

*Testing per Specification 4.4.6.2.2d. not applicable due to positive indication of valve position in Control Room.

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

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

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

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

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

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

At All Other Times:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
Chloride	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
Fluoride	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

*Limit not applicable with T_{avg} less than or equal to 250°F.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

*Not required with T_{avg} less than or equal to 250°F

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 but within the Allowable Limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. The provisions of Specification 3.0.4 are not applicable;

b. With the total cumulative operating time at a reactor coolant specific activity greater than 1 microCurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6-month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable;

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

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

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

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram of gross radioactivity, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits. In the next Annual Report, pursuant to Specification 6.9.1.4, submit the results of the specific activity analyses together with the following information:

a. Reactor power history starting 48 hours prior to the sample in which the limit was exceeded;

b. Results of: (1) the last isotopic analysis for radioiodines performed prior to exceeding the limit, (2) analysis while limit was exceeded, and (3) one analysis after the radioiodine activity was reduced to less than the limit, including for each isotopic analysis, the date and time of sampling and the radioiodine concentrations;

c. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded;

d. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and

e. The time duration when the specific activity of the reactor coolant exceeded 1 microCurie per gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

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

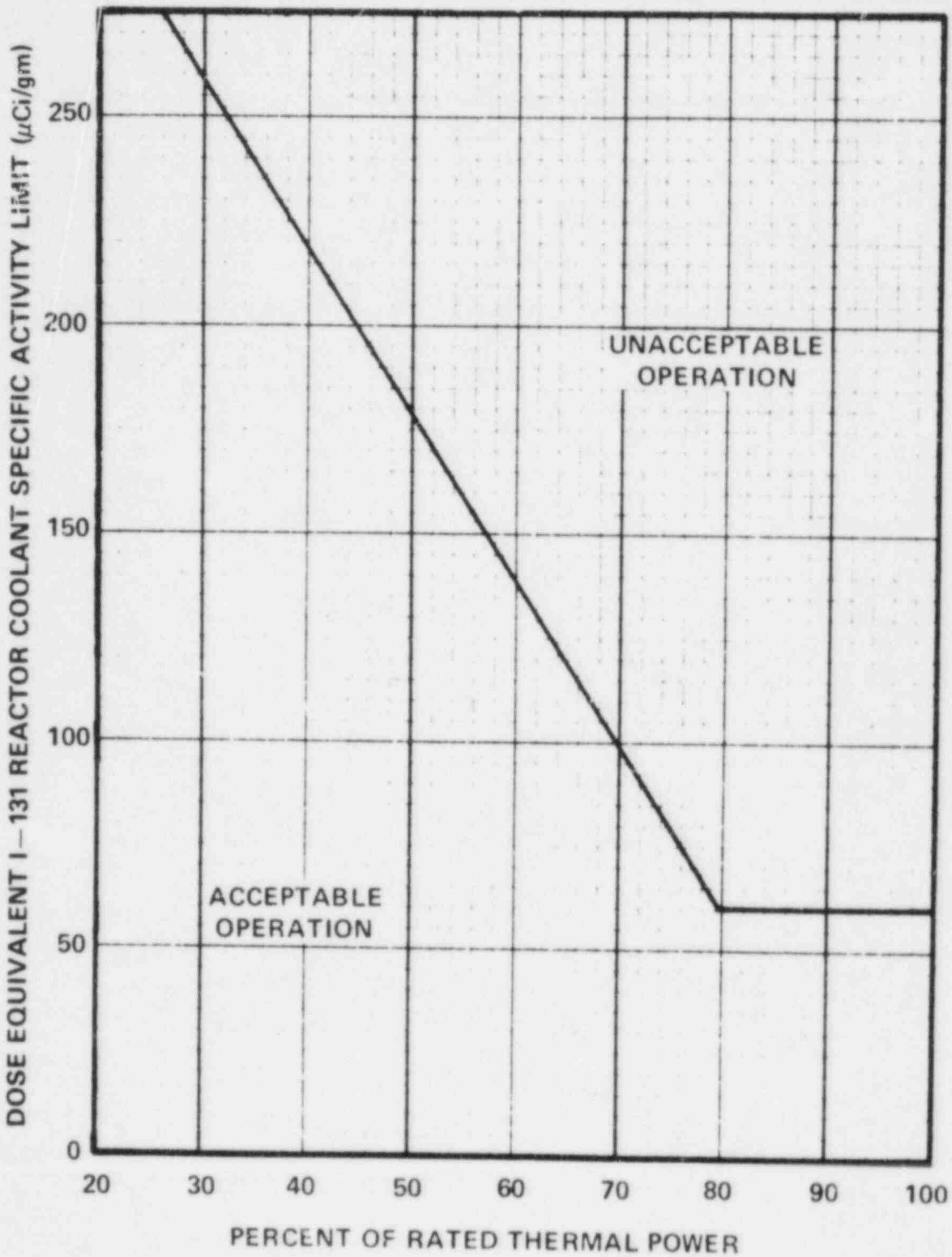

FIGURE 3.4-1

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

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination**	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for Ē Determination***	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT I-131 or 100/Ē μCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

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

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

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

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

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 T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

MATERIAL PROPERTY BASIS

COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT% (ACTUAL CONTENT = 0.08 WT%)

RT_{NDT} INITIAL : CONSERVATIVELY ASSUMED TO BE 40°F (ACTUAL RT_{NDT} = -8°F)

RT_{NDT} AFTER 16 EFPY : 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

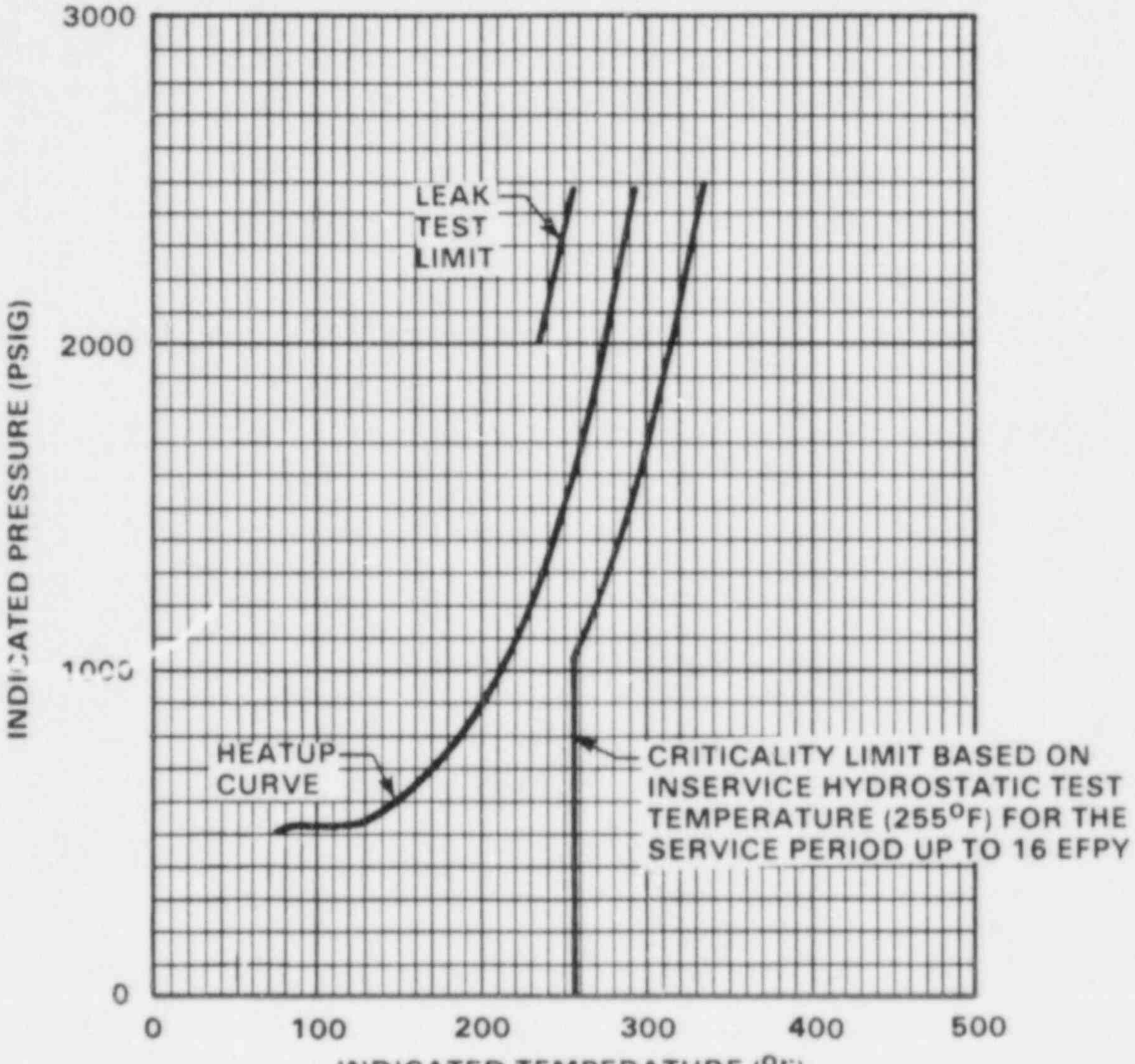

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 16 EFPY

MATERIAL PROPERTY BASIS

COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT% (ACTUAL CONTENT = 0.08 WT%)

RT_{NDT} INITIAL : CONSERVATIVELY ASSUMED TO BE 40°F (ACTUAL RT_{NDT} = -8°F)

RT_{NDT} AFTER 16 EFPY: 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

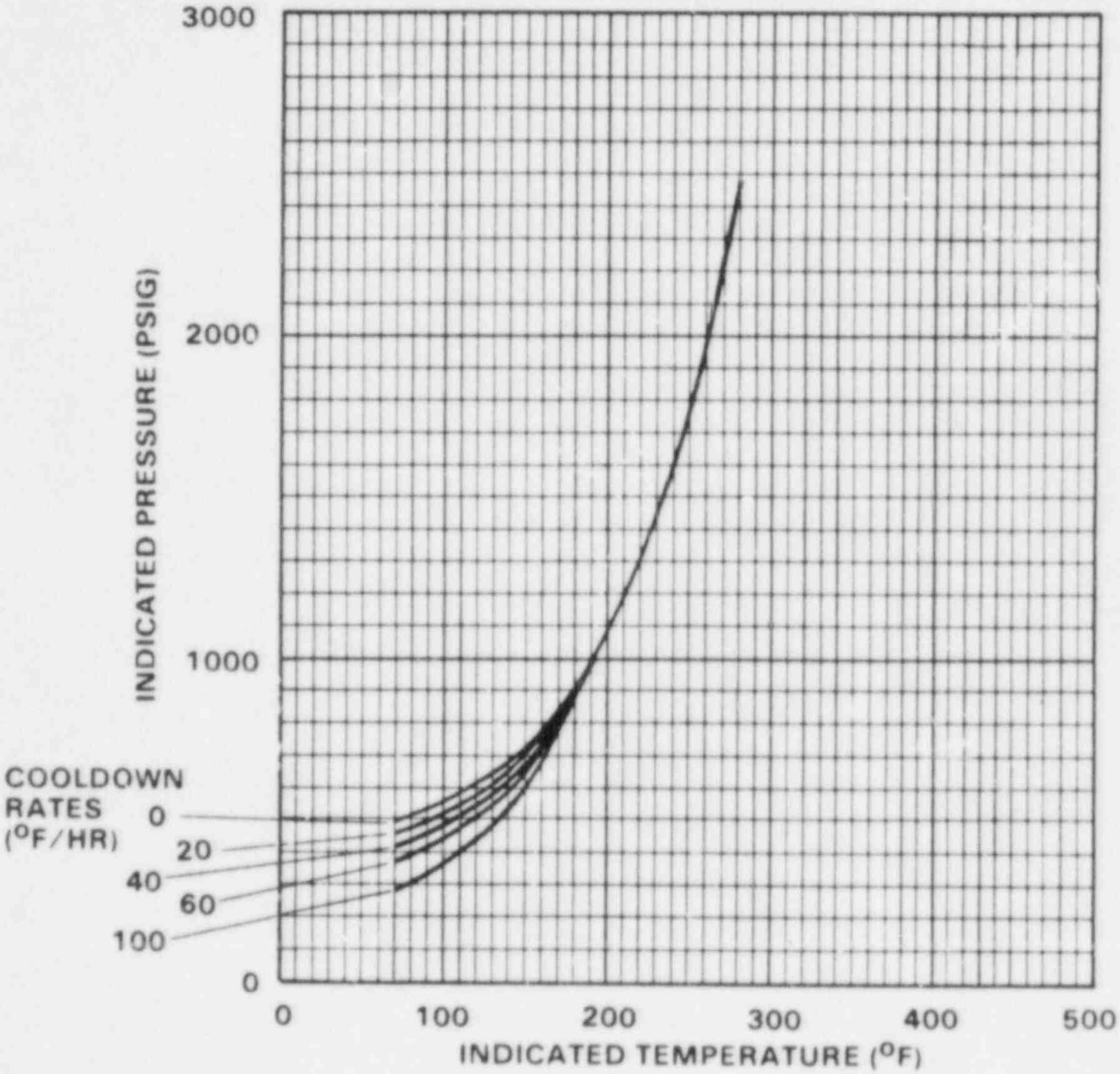

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 16 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	58.5°	4	First Refueling
V	61°	3.69	9
W	121.5°	4	Standby
X	238.5°	4	Standby
Y	241°	3.69	6
Z	301.5°	4	Standby

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of less than or equal to 450 psig, or

b. The Reactor Coolant System depressurized with a Reactor Coolant System vent of greater than or equal to 4.5 square inches.

APPLICABILITY: MODE 4 when the temperature of any Reactor Coolant System cold leg is less than or equal to 285°F, MODE 5 and MODE 6 with the reac or vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the Reactor Coolant System through at least a 4.5 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the Reactor Coolant System through at least a 4.5 square inch vent within 8 hours.

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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 least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The Reactor Coolant System vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

3/4 4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: All MODES.

ACTION:

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

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of at least two valves in series powered from emergency buses shall be OPERABLE and closed* at each of the following locations:

a. Reactor Vessel head

b. Pressurizer steam space

APPLICABILITY: Modes 1, 2, 3 and 4

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.11 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

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

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

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

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7853 and 8171 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 385 and 481 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2) Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 75 gallons by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the Reactor Coolant System pressure is above 2000 psig by verifying that power is removed from the isolation valve operators on Valves NI54A, NI65B, NI76A, and NI88B and that the respective circuit breakers are padlocked; and

d. At least once per 18 months by verifying that each cold leg injection accumulator isolation valve opens automatically under each of the following conditions:

   1) When an actual or a simulated Reactor Coolant System pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

   2) Upon receipt of a Safety Injection test signal.

4.5.1.1.2 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

UPPER HEAD INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each Upper Head Injection Accumulator System shall be OPERABLE with:

a. The discharge isolation valves open,

b. A minimum contained borated water volume of 1807 cubic feet,

c. A boron concentration of between 1900 and 2100 ppm, and

d. The nitrogen-bearing accumulator pressurized to between 1185 and 1285 psig.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With the Upper Head Injection Accumulator System inoperable, except as a result of closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water level in the surge tank and nitrogen pressure in the accumulators, and

2) Verifying that each accumulator discharge isolation valve is open.

---

*Pressurizer pressure above 1900 psig.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 138.3 gallons by verifying the boron concentration of the solution in the water-filled accumulator;

c. At least once per 18 months by:

   1) Verifying that each accumulator discharge isolation valve closes automatically when the water level is 93.2 ± 2.7 inches (Unit 1) and 93.1 ± 2.7 inches (Unit 2) above the working line on the water-filled accumulator, and

   2) Verifying that the total dissolved nitrogen and air in the water-filled accumulator is less than 80 scf per 1800 cubic feet of water (equivalent to 5×10^{-5} pound of nitrogen per pound of water).

d. At least once per 5 years and if the requirements of Specification 4.5.1.2c.2) are not met by replacing the membrane installed between the water-filled and nitrogen-bearing accumulators.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
NI-162A	Cold Leg Recirc.	Open
NI-121A	Hot Leg Recirc.	Closed
NI-152B	Hot Leg Recirc.	Closed
NI-183B	Hot Leg Recirc.	Closed
NI-173A	Residual Heat Removal Pump Disch.	Open
NI-178B	Residual Heat Removal Pump Disch.	Open
NI-100B	Safety Injection Pump Suction from Refueling Water Storage Tank	Open
NI-147B	Safety Injection Pump Mini-flow	Open

b. At least once per 31 days 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 least once per 18 months by:

1) Verifying automatic isolation and interlock action of the residual heat removal system from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 660 psig the interlocks will cause the valves to automatically close.

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

e. At least once per 18 months, during shutdown, by:

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

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

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) Residual heat removal pump.

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

1) Centrifugal charging pump $\geq$ 2380 psid,

2) Safety Injection pump $\geq$ 1430 psid, and

3) Residual heat removal pump $\geq$ 165 psid.

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

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

2) At least once per 18 months.

Centrifugal Charging Pump Injection Throttle Valve Number	Safety Injection Throttle Valve Number
NI-14	NI-164
NI-16	NI-166
NI-18	NI-168
NI-20	NI-170

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

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 333 gpm, and

b) The total pump flow rate is less than or equal to 565 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 462 gpm, and

b) The total pump flow rate is less than or equal to 660 gpm.

3) For residual heat removal pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3648 gpm.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A maximum of one centrifugal charging pump and one Safety Injection pump shall be OPERABLE whenever the temperature of one or more of the Reactor Coolant System cold legs is less than or equal to 285°F.

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All charging pumps and Safety Injection pumps, except the above required OPERABLE centrifugal charging pump, shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position or the discharge of each pump has been isolated from the Reactor Coolant System by at least two isolation valves with power removed from the valve motor operators at least once per 12 hours whenever the temperature of one or more of the Reactor Coolant System cold legs is less than or equal to 285°F.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum contained borated water volume of 363,513 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron,

c. A minimum solution temperature of 70°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The refueling water storage tank shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the refueling water storage tank temperature when the outside air temperature is less than 70°F or greater than 100°F.

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 least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3;

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

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 14.68 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than to 0.60 L_a.

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

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 14.68 psig, or

2) (Unit 1) Less than or equal to L_t, 0.122% by weight of the containment air per 24 hours at a reduced pressure of P_t, 7.34 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, and

c. A combined bypass leakage rate of less than 0.07 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

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

ACTION:

With: (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) the combined bypass leakage rate exceeding 0.07 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.07 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972 or the mass-plot method:

SURVEILLANCE REQUIREMENTS (Continued)

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a, 14.68 psig, or (Unit 1) at P_t, 7.34 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

b. If any periodic Type A test fails to meet either 0.75 L_a or (Unit 1) 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or (Unit 1) 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or (Unit 1) 0.75 L_t at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a or (Unit 1) 0.25 L_t;

   2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a or (Unit 1) 0.75 L_t and 1.25 L_t.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 14.68 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks,

   2) Purge supply and exhaust isolation valves with resilient material seals, and

   3) Dual-ply bellows assemblies on containment penetrations between the containment building and the annulus.

e. The combined bypass leakage rate shall be determined to be less than 0.07 L_a by applicable Type B and C tests at least once per 24 months except[a] for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 14.68 psig, or (Unit 1) P_t, 7.34 psig, during each Type A test;

SURVEILLANCE REQUIREMENTS (Continued)

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.9.3 or 4.6.1.9.4, as applicable;

g. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

h. The space between each dual-ply bellows assembly on containment penetrations between the containment building and the annulus shall be vented to the annulus during Type A tests. At least once per 24 months, the space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3 to 5 psig to verify no detectable leakage or the dual-ply bellows assembly shall be subjected to a leak test with the pressure on the containment side of the dual-ply bellows assembly at P_a, 14.68 psig, to verify the leakage to be within the limits of Specification 4.6.1.2e.; and

i. The provisions of Specification 4.0.2 are not applicable.

TABLE 3.6-1

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
M216	Pressurizer Relief Tank Makeup	Auxiliary Building	Type C
M212	Nitrogen to Pressurizer Relief Tank	Auxiliary Building	Type C
M327	Reactor Coolant Pump Motor Drain Tank Pump Discharge	Auxiliary Building	Type C
M259	Reactor Makeup Water Flush Header	Auxiliary Building	Type C
M373	Ice Condenser Glycol Pumps Discharge Line	Auxiliary Building	Type C
M372	Ice Condenser Glycol Pumps Suction Line	Auxiliary Building	Type C
M332	Cont. Hydrogen Purge Inlet Blower Discharge Line	Atmosphere	Type C
M348	Reactor Coolant Drain Tank Gas Space to Waste Gas System	Auxiliary Building	Type C
M221	Ventilation Unit Condensate Drain Header	Auxiliary Building	Type C
M356	Equipment Decontamination Line	Auxiliary Building	Type C
M358	Refueling Water Pump Suction	Auxiliary Building	Type C
M377	Refueling Cavity Fill Line	Auxiliary Building	Type C
M235	Pressurizer Sample	Auxiliary Building	Type C
M310	Reactor Coolant Hot Leg Sample	Auxiliary Building	Type C

TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
M323	Component Cooling to Component Cooling Drain Sump	Auxiliary Building	Type C
M240	Nuclear Service Water to Reactor Coolant Pump and Lower Cont. Vent. Units	Auxiliary Building	Type C
M230	Nuclear Service Water From Reactor Coolant Pump and Lower Cont. Vent. Units	Auxiliary Building	Type C
M385	Nuclear Service Water to Upper Containment Ventilation Units In	Turbine Building	Type C
M308	Nuclear Service Water to Upper Containment Ventilation Units Out	Turbine Building	Type C
M213	Incore Instrumentation Room Purge In	Auxiliary Building	Type C
M140	Incore Instrumentation Room Purge Out	Auxiliary Building	Type C
M456	Upper Compartment Purge Inlet	Auxiliary Building	Type C
M432	Upper Compartment Purge Inlet	Auxiliary Building	Type C
M357	Lower Compartment Purge Inlet	Auxiliary Building	Type C
M368	Containment Purge Exhaust	Auxiliary Building	Type C
M433	Containment Purge Exhaust	Auxiliary Building	Type C
M434	Lower Compartment Purge Inlet	Auxiliary Building	Type C

TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
M386	Containment Air Release	Auxiliary Building	Type C
M204	Containment Air Addition	Auxiliary Building	Type C
M316	Int. Fire Protection Header - Hose Racks	Auxiliary Building	Type C
M337	Demineralized Water	Auxiliary Building	Type C
M220	Instrument Air	Auxiliary Building	Type C
M219	Station Air	Auxiliary Building	Type C
M215	Breathing Air	Auxiliary Building	Type C
M329	Reactor Coolant Pump Motor Oil Fill	Auxiliary Building	Type C
M361	Int. Fire Protection Header - Sprinklers	Auxiliary Building	Type C
M119	Containment Purge Exhaust	Auxiliary Building	Type C
M331	Nitrogen Supply to Cold Leg Accumulators	Auxiliary Building	Type C
M322	Safety Injection Test Line	Auxiliary Building	Type C
M454	UHI Test Line	Auxiliary Building	Type C
M328*	Component Cooling to Reactor Vessel Support and RCP Coolers	Auxiliary Building	Type C

*Not applicable for Unit 1 until after the first refueling outage.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 14.68 psig.

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

ACTION:

a. With one containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of 14.68 psig;

b. By conducting overall air lock leakage tests at not less than P_a, 14.68 psig, and verifying the overall air lock leakage rate is within its limit:

   1) At least once per 6 months,# and

   2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

d. At least once per 6 months by conducting a pressure test at not less than P_a, 14.68 psig, to verify door seal integrity, with a measured leak rate of less than 15 sccm per door seal.

#The provisions of Specification 4.0.2 are not applicable.

*This represents an exemption to Appendix J of 10 CFR Part 50.

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 least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained:

a. Between 75* and 100°F in the containment upper compartment, and

b. Between 100* and 120°F in the containment lower compartment.

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

ACTION:

With the containment average air temperature not conforming to the above limits, restore the air temperature to within the limits 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

4.6.1.5.1 The primary containment upper compartment average air temperature shall be the arithmetical average of ambient air temperature monitoring stations located in the upper compartment. Temperature readings shall be obtained at least once per 24 hours from the elevation of 653 feet at the inlet of each operating upper containment ventilation unit.

4.6.1.5.2 The primary containment lower compartment average air temperature shall be the arithmetical average of ambient air temperature monitoring stations located in the lower compartment. Temperature readings shall be obtained at least once per 24 hours from the elevation of 570 feet at the inlet of each operating lower compartment ventilation unit.

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*Lower limit may be reduced to 60°F in MODE 2, 3 or 4.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission within 15 days as a Special Report pursuant to Specification 6.9.2.

REACTOR BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the reactor building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the reactor building shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the reactor building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the reactor building detected during the above required inspections shall be reported to the Commission within 15 days as a Special Report pursuant to Specification 6.9.2.

ANNULUS VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 Two independent Annulus Ventilation Systems shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.8 Each Annulus Ventilation System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and carbon adsorbers and verifying that the system operates for at least 10 continuous hours with the pre-heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or carbon adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) 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 9000 cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

   3) Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

*The requirement for reducing refrigerant concentration to 0.01 ppm may be satisfied by operating the system for 10 hours with heaters on and operating.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of carbon 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

d. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters, carbon adsorber banks, and moisture separators is less than 8 inches Water Gauge while operating the system at a flow rate of 9000 cfm ± 10%;

   2) Verifying that the system starts automatically on any Phase "A" Isolation test signal,

   3) Verifying that the filter cooling electric motor-operated bypass valves can be manually opened,

   4) Verifying that each system produces a negative pressure of greater than or equal to 0.5 inch Water Gauge in the annulus within 1 minute after a start signal, and

   5) Verifying that the pre-heaters dissipate 45 ± 6.7 kW.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 9000 cfm ± 10%; and

f. After each complete or partial replacement of a carbon adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9000 cfm ± 10%.

CONTAINMENT PURGE SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.1.9 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each containment purge supply and/or exhaust isolation valve for the lower compartment and the upper compartment (24-inch), instrument room (12-inch), and the Hydrogen Purge System (4-inch) shall be sealed closed, and

b. The Containment Air Release and Addition System (4-inch) isolation valve(s) may be open for up to 2000 hours during a calendar year.

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

ACTION:

a. With any containment purge supply and/or exhaust isolation valve for the lower compartment and the upper compartment, or instrument room, or Hydrogen Purge System open or not sealed closed, close and/or seal closed that valve or isolate the penetrations(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 Containment Air Release and Addition System isolation valve(s) open for more than 2000 hours during a calendar year, close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.9.3 and/or 4.6.1.9.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 Each containment purge supply and/or exhaust isolation valves for the lower compartment and the upper containment, or instrument room, or Hydrogen Purge System shall be verified to be sealed closed at least once per 31 days.

4.6.1.9.2 The cumulative time that the Containment Air Release and Addition System has been open during a calendar year shall be determined at least once per 7 days.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.9.3 At least once per 6 months on a STAGGERED TEST BASIS, the inboard and outboard valves with resilient material in each sealed closed containment purge supply and exhaust penetration for the lower compartment and the upper compartment, or instrument room, or Hydrogen Purge System shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.

4.6.1.9.4 At least once per 3 months the Containment Air Release and Addition System valves 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.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the refueling water storage tank and transferring suction to the containment sump.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2 Each Containment Spray System shall be demonstrated OPERABLE:

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

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 185 psid when tested pursuant to Specification 4.0.5;

c. At least once per 18 months during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Phase "B" Isolation test signal, and

   2) Verifying that each spray pump starts automatically on a Phase "B" Isolation test signal.

   3) Verifying that each spray pump is prevented from starting by the Containment Pressure Control System when the containment atmosphere pressure is less than or equal to 0.25 psid, and is allowed to start at greater than or equal to 0.45 psid relative to the outside atmosphere,

SURVEILLANCE REQUIREMENTS (Continued)

4) Verifying that each spray pump discharge valve closes or is prevented from opening by the Containment Pressure Control System when the containment atmosphere pressure is less than or equal to 0.25 psid and is allowed to open at greater than or equal to 0.45 psid relative to the outside atmosphere, and

5) Verifying that each spray pump is automatically deenergized by the Containment Pressure Control System when the containment atmosphere pressure is less than or equal to 0.25 psid relative to the outside atmosphere.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Tables 3.6-2a and 3.6-2b shall be OPERABLE with isolation times as shown in Tables 3.6-2a and 3.6-2b.

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

ACTION:

With one or more of the isolation valve(s) specified in Tables 3.6-2a and 3.6-2b inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

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

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Tables 3.6-2a and 3.6-2b 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.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve specified in Tables 3.6-2a and 3.6-2b shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position;

c. Verifying that on a Containment Radioactivity-High test signal, each purge and exhaust valve actuates to its isolation position; and

d. Verifying that on a High Relative Humidity ($\geq$70%) isolation test signal, each upper and lower containment purge supply and exhaust valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve of Tables 3.6-2a and 3.6-2b shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-2a

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation		
BB-57B#	Steam Generator 1A Blowdown Containment Outside Isolation	≤10
BB-21B#	Steam Generator 1B Blowdown Containment Outside Isolation	≤10
BB-61B#	Steam Generator 1C Blowdown Containment Outside Isolation	≤10
BB-10B#	Steam Generator 1D Blowdown Containment Outside Isolation	≤10
BB-56A#	Steam Generator 1A Blowdown Containment Inside Isolation	≤10
BB-19A#	Steam Generator 1B Blowdown Containment Inside Isolation	≤10
BB-60A#	Steam Generator 1C Blowdown Containment Inside Isolation	≤10
BB-8A#	Steam Generator 1D Blowdown Containment Inside Isolation	≤10
BB-148B#	Steam Generator 1A Blowdown Containment Isolation Bypass	≤10
BB-150B#	Steam Generator 1B Blowdown Containment Isolation Bypass	≤10
BB-149B#	Steam Generator 1C Blowdown Containment Isolation Bypass	≤10
BB-147B#	Steam Generator 1D Blowdown Containment Isolation Bypass	≤10
CA-149#	Steam Generator 1A Main Feedwater to Auxiliary Feedwater Nozzle Isolation	≤5
CA-150#	Steam Generator 1B Main Feedwater to Auxiliary Feedwater Nozzle Isolation	≤5
CA-151#	Steam Generator 1C Main Feedwater to Auxiliary Feedwater Nozzle Isolation	≤5
CA-152#	Steam Generator 1D Main Feedwater to Auxiliary Feedwater Nozzle Isolation	≤5
CA-185#	Auxiliary Nozzle Temper SG1A	≤5
CA-186#	Auxiliary Nozzle Temper SG1B	≤5
CA-187#	Auxiliary Nozzle Temper SG1C	≤5
CA-188#	Auxiliary Nozzle Temper SG1D	≤5
CF-60#	Steam Generator 1D Feedwater Containment Isolation	≤5
CF-51#	Steam Generator 1C Feedwater Containment Isolation	≤5
CF-42#	Steam Generator 1B Feedwater Containment Isolation	≤5
CF-33#	Steam Generator 1A Feedwater Containment Isolation	≤5
CF-90#	Steam Generator 1A Feedwater Purge Valve	≤5
CF-89#	Steam Generator 1B Feedwater Purge Valve	≤5
CF-88#	Steam Generator 1C Feedwater Purge Valve	≤5
CF-87#	Steam Generator 1D Feedwater Purge Valve	≤5

TABLE 3.6-2a (Continued)

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
KC-305B#	Excess Letdown Hx Supply Containment Isolation (Outside)	≤20
KC-315B#	Excess Letdown Hx Return Header Containment Isolation (Outside)	≤20
KC-320A#	NCDT Hx Supply Hdr Containment Isolation (Outside)	≤20
KC-332B#	NCDT Hx Return Hdr Containment Isolation (Inside)	≤20
KC-333A#	NCDT Hx Return Hdr Containment Isolation (Outside)	≤20
KC-429B	RB Drain Header Inside Containment Isolation	≤10
KC-430A	RB Drain Header Outside Containment Isolation	≤10
NB-260B	Reactor Makeup Water Tank to Flush Header	≤10
NC-53B	Nitrogen to Pressurizer Relief Tank #1 Containment Isolation Outside	≤10
NC-54A	Nitrogen to Pressurizer Relief Tank #1 Containment Isolation Inside	≤10
NC-56B	RMW Pump Disch Cont Isolation	≤10
NC-195B	NC Pump Motor Oil Containment Isolation Outside	≤10
NC-196A	NC Pump Motor Oil Containment Isolation Inside	≤10
NF-228A	Unit 1 Air Handling Units Glycol Supply Containment Isolation Outside	≤10
NF-233B	Unit 1 Air Handling Units Glycol Return Containment Isolation Inside	≤10
NF-234A	Unit 1 Air Handling Units Glycol Return Containment Isolation Outside	≤10
NI-47A	Accumulator N_2 Supply Outside Containment Isolation	≤10
NI-95A	Test Hdr Inside Containment Isolation	≤10
NI-96B	Test Hdr Outside Containment Isolation	≤10
NI-120B	Safety Injection Pump to Accumulator Fill Line Isolation	≤10
NI-122B#	Hot Leg Injection Check 1NI124, 1NI128 Test Isolation	≤10
NI-154B#	Hot Leg Recirculation Check 1NI125, 1NI129 Test Isolation	≤10
NI-255B	UHI Check Valve Test Line Isolation	≤10
NI-258A	UHI Check Valve Test Line Isolation	≤10
NI-264B	UHI Check Valve Test Line Outside Containment Isolation	≤10

TABLE 3.6-2a (Continued)

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
NI-266A	UHI Check Valve Test Line Inside Containment Isolation	≤10
NI-267A	UHI Check Valve Test Line Inside Containment Isolation	≤10
NI-153A#	Hot Leg Injection Check NI156, NI159 Test Isolation	≤10
NM-3A	Pressurizer Liquid Sample Line Inside Containment Isolation	≤10
NM-6A	Pressurizer Steam Sample Line Inside Containment Isolation	≤10
NM-7B	Pressurizer Sample Header Outside Containment Isolation	≤10
NM-22A	NC Hot Leg A Sample Line Inside Containment Isolation	≤10
NM-25A	NC Hot Leg C Sample Line Inside Containment Isolation	≤10
NM-26B	NC Hot Leg Sample Hdr Outside Containment Isolation	≤10
NM-72B	NI Accumulator 1A Sample Line Inside Containment Isolation	≤10
NM-75B	NI Accumulator 1B Sample Line Inside Containment Isolation	≤10
NM-78B	NI Accumulator 1C Sample Line Inside Containment Isolation	≤10
NM-81B	NI Accumulator 1D Sample Line Inside Containment Isolation	≤10
NM-82A	NI Accumulator Sample Hdr Outside Containment Isolation	≤10
NM-187A#	SG 1A Upper Shell Sample Containment Isolation Inside	≤10
NM-190A#	SG 1A Blowdown Line Sample Containment Isolation Inside	≤10
NM-191B#	SG 1A Sample Hdr Containment Isolation Outside	≤10
NM-197B#	SG 1B Upper Shell Sample Containment Isolation Inside	≤10
NM-200B#	SG 1B Blowdown Line Sample Containment Isolation Inside	≤10
NM-201A#	SG 1B Sample Hdr Containment Isolation Outside	≤10
NM-207A#	SG 1C Upper Shell Sample Containment Isolation Inside	≤10
NM-210A#	SG 1C Blowdown Line Sample Containment Isolation Inside	≤10
NM-211B#	SG 1C Sample Hdr Containment Isolation Outside	≤10
NM-217B#	SG 1D Upper Shell Sample Containment Isolation Inside	≤10
NM-220B#	SG 1D Blowdown Line Sample Containment Isolation Inside	≤10
NM-221A#	SG 1D Sample Hdr Containment Isolation Outside	≤10
NV-15B	Letdown Containment Isolation Outside	≤10
NV-89A	NC Pumps Seal Return Containment Isolation Inside	≤10
NV-91B	NC Pumps Seal Return Containment Isolation Outside	≤10
NV-314B#	Charging Line Containment Isolation Outside	≤10

TABLE 3.6-2a (Continued)

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
NV-11A	45 gpm Letdown Orifice Outlet - Containment Isolation	≤10
NV-13A	75 gpm Letdown Orifice Outlet - Containment Isolation	≤10
NV-10A	High Pressurizer Letdown Orifice Outlet - Containment Isolation	≤10
NV-872A	Standby Makeup Pump to RCS seals	≤10
RF-389B	Interior Fire Protection Containment Hose Rack Isolation Valve (Outside Containment)	≤5
RF-447B	Reactor Building Sprinklers Containment Isolation Valve (Outside Containment)	≤5
VB-83B	Breathing Air Unit 1 Containment Isolation	≤10
VY-18B**	Containment H_2 Purge to Annulus Inside Containment Isolation	≤10
VY-17A**	Containment H_2 Purge to Annulus Outside Containment Isolation	≤10
VY-15B**	Containment H_2 Purge Blower Outlet, Containment Isolation (Outside)	≤10
VI-312A	RB Isolation Valve for VI Supply to annulus Vent.	≤10
VP-1B**	Upper Containment Purge Supply #1 Outside Isolation	≤5
VP-2A**	Upper Containment Purge Supply #1 Inside Isolation	≤5
VP-3B**	Upper Containment Purge Supply #2 Outside Isolation	≤5
VP-4A**	Upper Containment Purge Supply #2 Inside Isolation	≤5
VP-6B**	Lower Containment Purge Supply #1 Outside Isolation	≤5
VP-7A**	Lower Containment Purge Supply #1 Inside Isolation	≤5
VP-8B**	Lower Containment Purge Supply #2 Outside Isolation	≤5
VP-9A**	Lower Containment Purge Supply #2 Inside Isolation	≤5
VP-10A**	Upper Containment Purge Exhaust #1 Inside Isolation	≤5

TABLE 3.6-2a (Continued)

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
VP-11B**	Upper Containment Purge Exhaust #1 Outside Isolation	≤5
VP-12A**	Upper Containment Purge Exhaust #2 Inside Isolation	≤5
VP-13B**	Upper Containment Purge Exhaust #2 Outside Isolation	≤5
VP-15A**	Lower Containment Purge Exhaust #1 Inside Isolation	≤5
VP-16B**	Lower Containment Purge Exhaust #1 Outside Isolation	≤5
VP-17A**	Incore Instru. Room Purge Supply Inside Isolation	≤5
VP-18B**	Incore Instru. Room Purge Supply Outside Isolation	≤5
VP-19A**	Incore Instru. Room Purge Exhaust Inside Isolation	≤5
VP-20B**	Incore Instru. Room Purge Exhaust Outside Isolation	≤5
VQ-2A**	Containment Air Release Inside Isolation	≤5
VQ-3B**	Containment Air Release Outside Isolation	≤5
VQ-15B**	Containment Air Addition Outside Isolation	≤5
VQ-16A**	Containment Air Addition Inside Isolation	≤5
VS-54B	Unit 1 Containment Header Outside Isolation	≤15
WL-807B#	NCDT Pumps Discharge Outside Containment Isolation	≤10
WL-805A#	NCDT Pumps Discharge Inside Containment Isolation	≤10
WL-450A	NCDT Vent Inside Containment Isolation	≤10
WL-451B	NCDT Vent Outside Containment Isolation	≤10
WL-825A#**	RB Sump Pump Discharge Inside Containment Isolation	≤10
WL-827B#**	RB Sump Pump Discharge Outside Containment Isolation	≤10
YM-119B	Demin. Water Containment Outside Isolation	≤10
2. Phase "B" Isolation		
KC-338B#	NC Pump Supply Header Pent. Isolation (Outside)	≤40
KC-424B#	NC Pumps Return Hdr. Pent. Inside Isolation	≤40
KC-425A#	NC Pumps Return Hdr. Outside Isolation	≤40

TABLE 3.6-2a (Continued)

UNTI 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
2. Phase "B" Isolation (Continued)		
RN-437B	Supply to NC Pumps and LCVU Supply Outside Containment Isolation	≤60
RN-484A	Return from NC Pumps and LCVU Return Inside Containment Isolation	≤60
RN-487B	Return from NC Pumps and LCVU Return Outside Containment Isolation	≤60
RN-404B	Supply to Upper Containment Supply Ventilation Units Containment Isolation (Outside)	≤10
RN-429A	Return from Upper Containment Ventilation Units Containment Isolation (Inside)	≤10
RN-432B	Return from Upper Containment Ventilation Units Containment Isolation (Outside)	≤10
VI-77B	Instrument Air Containment Outside Isolation	≤10
SM-1 #	Main Steam 1D Isolation	≤5
SM-3 #	Main Steam 1C Isolation	≤5
SM-5 #	Main Steam 1B Isolation	≤5
SM-7 #	Main Steam 1A Isolation	≤5
SM-9 #	Main Steam 1D Isolation Bypass Ctrl.	≤5
SM-10 #	Main Steam 1C Isolation Bypass Ctrl.	≤5
SM-11 #	Main Steam 1B Isolation Bypass Ctrl.	≤5
SM-12 #	Main Steam 1A Isolation Bypass Ctrl.	≤5
SV-19 #	Main Steam 1A PORV	≤5
SV-13 #	Main Steam 1B PORV	≤5
SV-7 #	Main Steam 1C PORV	≤5
SV-1 #	Main Steam 1D PORV	≤5
WL-867A**	Containment Vent Unit Drains Inside Containment Isolation	≤10
WL-869B**	Containment Vent Unit Drains Outside Containment Isolation	≤10

TABLE 3.6-2a (Continued)

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
3. Manual		
NC-141	NC Pump H_2 Drain Tank Pump Discharge	N.A.
NC-142	NC Pump H_2 Drain Tank Pump Discharge	N.A.
NI-3	Boron Injection Tank Line to Cold Legs	N.A.
FW-11	Refueling Water Pump Suction	N.A.
FW-13	Refueling Water Pump Suction	N.A.
CF-91#	Feedwater 1A	N.A.
CF-93#	Feedwater 1B	N.A.
CF-95#	Feedwater 1C	N.A.
CF-97#	Feedwater 1D	N.A.
CA-121#	Aux. Feedwater 1A	N.A.
BW-1#	Aux. Feedwater 1A	N.A.
CA-120#	Aux. Feedwater 1B	N.A.
BW-26#	Aux. Feedwater 1B	N.A.
CA-119#	Aux. Feedwater 1C	N.A.
BW-17#	Aux. Feedwater 1C	N.A.
CA-118#	Aux. Feedwater 1D	N.A.
BW-10#	Aux. Feedwater 1D	N.A.
SM-16#	Main Steam 1A	N.A.
SM-73#*	Main Steam 1A	N.A.
SM-105#	Main Steam 1A	N.A.
SM-121#	Main Steam 1A	N.A.
SM-143#	Main Steam 1A	N.A.
SM-72#*	Main Steam 1B	N.A.
SM-104#	Main Steam 1B	N.A.
SM-120#	Main Steam 1B	N.A.
SM-142#	Main Steam 1B	N.A.
SM-1#	Main Steam 1B	N.A.
SM-17#	Main Steam 1B	N.A.
SM-18#	Main Steam 1C	N.A.
SM-71#*	Main Steam 1C	N.A.

TABLE 3.6-2a (Continued)

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
3. Manual (Continued)		
SM-103#	Main Steam 1C	N.A.
SM-119#	Main Steam 1C	N.A.
SM-141#	Main Steam 1C	N.A.
SA-4#	Main Steam 1C	N.A.
SM-19#	Main Steam 1D	N.A.
SM-70#*	Main Steam 1D	N.A.
SM-102#	Main Steam 1D	N.A.
SM-118#	Main Steam 1D	N.A.
SM-140#	Main Steam 1D	N.A.
WE-20*	Cont Bldg Supply Isol	N.A.
WE-22*	Cont Bldg Supply Isol	N.A.
WE-56*	Cont Bldg Supply Isol	N.A.
FW-4*	Refueling Water	N.A.
NV-862#*	Pressurizer Auxiliary Spray ND Outside Containment	N.A.
WLA-21#*	Steam Generator Drain Pump Discharge Outside Containment Isolation	N.A.
WLA-24#*	Steam Generator Drain Pump Discharge Outside Containment Isolation	N.A.

TABLE NOTATIONS

* May be opened on an intermittent basis under administrative control.

** Valve also receives a High Radiation (H) and/or a High Relative Humidity isolation signal.

\# Not subject to Type C leakage tests.

NOTE: Times are for valve operation only, and do not include any sensor response or circuit delay times. See Specification 3/4 3.2 for system actuation response times.

TABLE 3.6-2b

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation		
BB-57B#	Steam Generator 2A Blowdown Containment Outside Isolation	≤10
BB-21B#	Steam Generator 2B Blowdown Containment Outside Isolation	≤10
BB-61B#	Steam Generator 2C Blowdown Containment Outside Isolation	≤10
BB-10B#	Steam Generator 2D Blowdown Containment Outside Isolation	≤10
BB-56A#	Steam Generator 2A Blowdown Containment Inside Isolation	≤10
BB-19A#	Steam Generator 2B Blowdown Containment Inside Isolation	≤10
BB-60A#	Steam Generator 2C Blowdown Containment Inside Isolation	≤10
BB-8A#	Steam Generator 2D Blowdown Containment Inside Isolation	≤10
BB-148B#	Steam Generator 2A Blowdown Containment Isolation Bypass	≤10
BB-150B#	Steam Generator 2B Blowdown Containment Isolation Bypass	≤10
BB-149B#	Steam Generator 2C Blowdown Containment Isolation Bypass	≤10
BB-147B#	Steam Generator 2D Blowdown Containment Isolation Bypass	≤10
CA-149#	Steam Generator 2A Main Feedwater to Auxiliary Feedwater Nozzle Isolation	≤5
CA-150#	Steam Generator 2B Main Feedwater to Auxiliary Feedwater Nozzle Isolation	≤5
CA-151#	Steam Generator 2C Main Feedwater to Auxiliary Feedwater Nozzle Isolation	≤5
CA-152#	Steam Generator 2D Main Feedwater to Auxiliary Feedwater Nozzle Isolation	≤5
CA-185#	Auxiliary Nozzle Temper SG2A	≤5
CA-186#	Auxiliary Nozzle Temper SG2B	≤5
CA-187#	Auxiliary Nozzle Temper SG2C	≤5
CA-188#	Auxiliary Nozzle Temper SG2D	≤5
CF-60#	Steam Generator 2D Feedwater Containment Isolation	≤5
CF-51#	Steam Generator 2C Feedwater Containment Isolation	≤5
CF-42#	Steam Generator 2B Feedwater Containment Isolation	≤5
CF-33#	Steam Generator 2A Feedwater Containment Isolation	≤5
CF-90#	Steam Generator 2A Feedwater Purge Valve	≤5
CF-89#	Steam Generator 2B Feedwater Purge Valve	≤5
CF-88#	Steam Generator 2C Feedwater Purge Valve	≤5
CF-87#	Steam Generator 2D Feedwater Purge Valve	≤5

TABLE 3.6-2b (Continued)

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
KC-305B#	Excess Letdown Hx Supply Containment Isolation (Outside)	≤20
KC-315B#	Excess Letdown Hx Return Header Containment Isolation (Outside)	≤20
KC-320A#	NCDT Hx Supply Hdr Containment Isolation (Outside)	≤20
KC-332B#	NCDT Hx Return Hdr Containment Isolation (Inside)	≤20
KC-333A#	NCDT Hx Return Hdr Containment Isolation (Outside)	≤20
KC-429B	RB Drain Header Inside Containment Isolation	≤10
KC-430A	RB Drain Header Outside Containment Isolation	≤10
NB-260B	Reactor Makeup Water Tank to Flush Header	≤10
NC-53B	Nitrogen to Pressurizer Relief Tank #1 Containment Isolation Outside	≤10
NC-54A	Nitrogen to Pressurizer Relief Tank #1 Containment Isolation Inside	≤10
NC-56B	RMW Pump Disch Cont Isolation	≤10
NC-195B	NC Pump Motor Oil Containment Isolation Outside	≤10
NC-196A	NC Pump Motor Oil Containment Isolation Inside	≤10
NF-228A	Unit 2 Air Handling Units Glycol Supply Containment Isolation Outside	≤10
NF-233B	Unit 2 Air Handling Units Glycol Return Containment Isolation Inside	≤10
NF-234A	Unit 2 Air Handling Units Glycol Return Containment Isolation Outside	≤10
NI-47A	Accumulator N_2 Supply Outside Containment Isolation	≤10
NI-95A	Test Hdr Inside Containment Isolation	≤10
NI-96B	Test Hdr Outside Containment Isolation	≤10
NI-120B	Safety Injection Pump to Accumulator Fill Line Isolation	≤10
NI-122B#	Hot Leg Injection Check 2NI124, 2NI128 Test Isolation	≤10
NI-154B#	Hot Leg Recirculation Check 2NI125, 2NI129 Test Isolation	≤10
NI-255B	UHI Check Valve Test Line Isolation	≤10
NI-258A	UHI Check Valve Test Line Isolation	≤10
NI-264B	UHI Check Valve Test Line Outside Containment Isolation	≤10

TABLE 3.6-2b (Continued)

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
NI-266A	UHI Check Valve Test Line Inside Containment Isolation	≤10
NI-267A	UHI Check Valve Test Line Inside Containment Isolation	≤10
NI-153A#	Hot Leg Injection Check NI156, NI159 Test Isolation	≤10
NM-3A	Pressurizer Liquid Sample Line Inside Containment Isolation	≤10
NM-6A	Pressurizer Steam Sample Line Inside Containment Isolation	≤10
NM-7B	Pressurizer Sample Header Outside Containment Isolation	≤10
NM-22A	NC Hot Leg A Sample Line Inside Containment Isolation	≤10
NM-25A	NC Hot Leg C Sample Line Inside Containment Isolation	≤10
NM-26B	NC Hot Leg Sample Hdr Outside Containment Isolation	≤10
NM-72B	NI Accumulator 2A Sample Line Inside Containment Isolation	≤10
NM-75B	NI Accumulator 2B Sample Line Inside Containment Isolation	≤10
NM-78B	NI Accumulator 2C Sample Line Inside Containment Isolation	≤10
NM-81B	NI Accumulator 2D Sample Line Inside Containment Isolation	≤10
NM-82A	NI Accumulator Sample Hdr Outside Containment Isolation	≤10
NM-187A#	SG 2A Upper Shell Sample Containment Isolation Inside	≤10
NM-190A#	SG 2A Blowdown Line Sample Containment Isolation Inside	≤10
NM-191B#	SG 2A Sample Hdr Containment Isolation Outside	≤10
NM-197B#	SG 2B Upper Shell Sample Containment Isolation Inside	≤10
NM-200B#	SG 2B Blowdown Line Sample Containment Isolation Inside	≤10
NM-201A#	SG 2B Sample Hdr Containment Isolation Outside	≤10
NM-207A#	SG 2C Upper Shell Sample Containment Isolation Inside	≤10
NM-210A#	SG 2C Blowdown Line Sample Containment Isolation Inside	≤10
NM-211B#	SG 2C Sample Hdr Containment Isolation Outside	≤10
NM-217B#	SG 2D Upper Shell Sample Containment Isolation Inside	≤10
NM-220B#	SG 2D Blowdown Line Sample Containment Isolation Inside	≤10
NM-221A#	SG 2D Sample Hdr Containment Isolation Outside	≤10
NV-15B	Letdown Containment Isolation Outside	≤10
NV-89A	NC Pumps Seal Return Containment Isolation Inside	≤10
NV-91B	NC Pumps Seal Return Containment Isolation Outside	≤10
NV-314B#	Charging Line Containment Isolation Outside	≤10

TABLE 3.6-2b (Continued)

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
NV-11A	45 gpm Letdown Orifice Outlet - Containment Isolation	≤10
NV-13A	75 gpm Letdown Orifice Outlet - Containment Isolation	≤10
NV-10A	High Pressurizer Letdown Orifice Outlet - Containment Isolation	≤10
NV-872A	Standby Makeup Pump to RCS seals	≤10
RF-389B	Interior Fire Protection Containment Hose Rack Isolation Valve (Outside Containment)	≤5
RF-447B	Reactor Building Sprinklers Containment Isolation Valve (Outside Containment)	≤5
VB-83B	Breathing Air Unit 2 Containment Isolation	≤10
VY-18B**	Containment H_2 Purge to Annulus Inside Containment Isolation	≤10
VY-17A**	Containment H_2 Purge to Annulus Outside Containment Isolation	≤10
VY-15B**	Containment H_2 Purge Blower Outlet, Containment Isolation (Outside)	≤10
VI-312A	RB Isolation Valve for VI Supply to annulus Vent.	≤10
VP-1B**	Upper Containment Purge Supply #1 Outside Isolation	≤5
VP-2A**	Upper Containment Purge Supply #1 Inside Isolation	≤5
VP-3B**	Upper Containment Purge Supply #2 Outside Isolation	≤5
VP-4A**	Upper Containment Purge Supply #2 Inside Isolation	≤5
VP-6B**	Lower Containment Purge Supply #1 Outside Isolation	≤5
VP-7A**	Lower Containment Purge Supply #1 Inside Isolation	≤5
VP-8B**	Lower Containment Purge Supply #2 Outside Isolation	≤5
VP-9A**	Lower Containment Purge Supply #2 Inside Isolation	≤5
VP-10A**	Upper Containment Purge Exhaust #1 Inside Isolation	≤5

TABLE 3.6-2b (Continued)

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
VP-11B**	Upper Containment Purge Exhaust #1 Outside Isolation	≤5
VP-12A**	Upper Containment Purge Exhaust #2 Inside Isolation	≤5
VP-13B**	Upper Containment Purge Exhaust #2 Outside Isolation	≤5
VP-15A**	Lower Containment Purge Exhaust #1 Inside Isolation	≤5
VP-16B**	Lower Containment Purge Exhaust #1 Outside Isolation	≤5
VP-17A**	Incore Instru. Room Purge Supply Inside Isolation	≤5
VP-18B**	Incore Instru. Room Purge Supply Outside Isolation	≤5
VP-19A**	Incore Instru. Room Purge Exhaust Inside Isolation	≤5
VP-20B**	Incore Instru. Room Purge Exhaust Outside Isolation	≤5
VQ-2A**	Containment Air Release Inside Isolation	≤5
VQ-3B**	Containment Air Release Outside Isolation	≤5
VQ-15B**	Containment Air Addition Outside Isolation	≤5
VQ-16A**	Containment Air Addition Inside Isolation	≤5
VS-54B	Unit 2 Containment Header Outside Isolation	≤15
WL-807B#	NCDT Pumps Discharge Outside Containment Isolation	≤10
WL-805A#	NCDT Pumps Discharge Inside Containment Isolation	≤10
WL-450A	NCDT Vent Inside Containment Isolation	≤10
WL-451B	NCDT Vent Outside Containment Isolation	≤10
WL-825A#**	RB Sump Pump Discharge Inside Containment Isolation	≤10
WL-827B#**	RB Sump Pump Discharge Outside Containment Isolation	≤10
YM-119B	Demin. Water Containment Outside Isolation	≤10
2. Phase "B" Isolation		
KC-338B#	NC Pump Supply Header Pent. Isolation (Outside)	≤40
KC-424B#	NC Pumps Return Hdr. Pent. Inside Isolation	≤40
KC-425A#	NC Pumps Return Hdr. Outside Isolation	≤40

TABLE 3.6-2b (Continued)

UNTI 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
2. Phase "B" Isolation (Continued)		
RN-437B	Supply to NC Pumps and LCVU Supply Outside Containment Isolation	≤ 60
RN-484A	Return from NC Pumps and LCVU Return Inside Containment Isolation	≤ 60
RN-487B	Return from NC Pumps and LCVU Return Outside Containment Isolation	≤ 60
RN-404B	Supply to Upper Containment Supply Ventilation Units Containment Isolation (Outside)	≤ 10
RN-429A	Return from Upper Containment Ventilation Units Containment Isolation (Inside)	≤ 10
RN-432B	Return from Upper Containment Ventilation Units Containment Isolation (Outside)	≤ 10
VI-77B	Instrument Air Containment Outside Isolation	≤ 10
SM-1 #	Main Steam 2D Isolation	≤ 5
SM-3 #	Main Steam 2C Isolation	≤ 5
SM-5 #	Main Steam 2B Isolation	≤ 5
SM-7 #	Main Steam 2A Isolation	≤ 5
SM-9 #	Main Steam 2D Isolation Bypass Ctrl.	≤ 5
SM-10 #	Main Steam 2C Isolation Bypass Ctrl.	≤ 5
SM-11 #	Main Steam 2B Isolation Bypass Ctrl.	≤ 5
SM-12 #	Main Steam 2A Isolation Bypass Ctrl.	≤ 5
SV-19 #	Main Steam 2A PORV	≤ 5
SV-13 #	Main Steam 2B PORV	≤ 5
SV-7 #	Main Steam 2C PORV	≤ 5
SV-1 #	Main Steam 2D PORV	≤ 5
WL-867A**	Containment Vent Unit Drains Inside Containment Isolation	≤ 10
WL-869B**	Containment Vent Unit Drains Outside Containment Isolation	≤ 10

TABLE 3.6-2b (Continued)

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
3. Manual		
NC-141	NC Pump H_2 Drain Tank Pump Discharge	N.A.
NC-142	NC Pump H_2 Drain Tank Pump Discharge	N.A.
NI-3	Boron Injection Tank Line to Cold Legs	N.A.
FW-11	Refueling Water Pump Suction	N.A.
FW-13	Refueling Water Pump Suction	N.A.
CF-91#	Feedwater 2A	N.A.
CF-93#	Feedwater 2B	N.A.
CF-95#	Feedwater 2C	N.A.
CF-97#	Feedwater 2D	N.A.
CA-121#	Aux. Feedwater 2A	N.A.
BW-1#	Aux. Feedwater 2A	N.A.
CA-120#	Aux. Feedwater 2B	N.A.
BW-26#	Aux. Feedwater 2B	N.A.
CA-119#	Aux. Feedwater 2C	N.A.
BW-17#	Aux. Feedwater 2C	N.A.
CA-118#	Aux. Feedwater 2D	N.A.
BW-10#	Aux. Feedwater 2D	N.A.
SM-16#	Main Steam 2A	N.A.
SM-73#*	Main Steam 2A	N.A.
SM-105#	Main Steam 2A	N.A.
SM-121#	Main Steam 2A	N.A.
SM-143#	Main Steam 2A	N.A.
SM-72#*	Main Steam 2B	N.A.
SM-104#	Main Steam 2B	N.A.
SM-120#	Main Steam 2B	N.A.
SM-142#	Main Steam 2B	N.A.
SM-1#	Main Steam 2B	N.A.
SM-17#	Main Steam 2B	N.A.
SM-18#	Main Steam 2C	N.A.
SM-71#*	Main Steam 2C	N.A.

TABLE 3.6-2b (Continued)

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
3. Manual (Continued)		
SM-103#	Main Steam 2C	N.A.
SM-119#	Main Steam 2C	N.A.
SM-141#	Main Steam 2C	N.A.
SA-4#	Main Steam 2C	N.A.
SM-19#	Main Steam 2D	N.A.
SM-70#*	Main Steam 2D	N.A.
SM-102#	Main Steam 2D	N.A.
SM-118#	Main Steam 2D	N.A.
SM-140#	Main Steam 2D	N.A.
WE-20*	Cont Bldg Supply Isol	N.A.
WE-22*	Cont Bldg Supply Isol	N.A.
WE-56*	Cont Bldg Supply Isol	N.A.
FW-4*	Refueling Water	N.A.
NV-862#*	Pressurizer Auxiliary Spray ND Outside Containment	N.A.
WLA-21#*	Steam Generator Drain Pump Discharge Outside Containment Isolation	N.A.
WLA-24#*	Steam Generator Drain Pump Discharge Outside Containment Isolation	N.A.

TABLE NOTATIONS

* May be opened on an intermittent basis under administrative control.

** Valve also receives a High Radiation (H) and/or a High Relative Humidity isolation signal.

Not subject to Type C leakage tests.

NOTE: Times are for valve operation only, and do not include any sensor response or circuit delay times. See Specification 3/4 3.2 for system actuation response times.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using hydrogen gas mixtures to obtain calibration points of:

a. One volume percent hydrogen, and

b. Four volume percent hydrogen.

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a Recombiner System functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW; and

b. At least once per 18 months by:

   1) Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

   2) Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

   3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

HYDROGEN MITIGATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 Both trains of the Hydrogen Mitigation System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one train of the Hydrogen Mitigation System inoperable, restore the inoperable train to OPERABLE status within 7 days or increase the surveillance interval of Specification 4.6.4.3a. from 92 days to 7 days on the OPERABLE train until the inoperable train is returned to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each train of the Hydrogen Mitigation System shall be demonstrated OPERABLE:

a. At least once per 92 days by energizing the supply breakers and verifying that at least 35 of 36 igniters are energized,* and

b. At least once per 18 months by verifying the temperature of each igniter is a minimum of 1700°F.

*Inoperable igniters must not be on corresponding redundant circuits which provide coverage for the same region.

3/4.6.5 ICE CONDENSER

ICE BED

LIMITING CONDITION FOR OPERATION

3.6.5.1 The ice bed shall be OPERABLE with:

a. The stored ice having a boron concentration of at least 1800 ppm boron as sodium tetraborate and a pH of 9.0 to 9.5,

b. Flow channels through the ice condenser,

c. A maximum ice bed temperature of less than or equal to 27°F,

d. A total ice weight of at least 2,368,652 pounds at a 95% level of confidence, and

e. 1944 ice baskets.

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

ACTION:

With the ice bed inoperable, restore the ice bed to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 The ice condenser shall be determined OPERABLE:

a. At least once per 12 hours by using the Ice Bed Temperature Monitoring System to verify that the maximum ice bed temperature is less than or equal to 27°F,

b. At least once per 9 months by:

1) Chemical analyses which verify that at least nine representative samples of stored ice have a boron concentration of at least 1800 ppm as sodium tetraborate and a pH of 9.0 to 9.5 at 25°C;

2) Weighing a representative sample of at least 144 ice baskets and verifying that each basket contains at least 1218 lbs of ice. The representative sample shall include six baskets from each of the 24 ice condenser bays and shall be constituted of

SURVEILLANCE REQUIREMENTS (Continued)

one basket each from Radial Rows 1, 2, 4, 6, 8, and 9 (or from the same row of an adjacent bay if a basket from a designated row cannot be obtained for weighing) within each bay. If any basket is found to contain less than 1218 pounds of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The minimum average weight of ice from the 20 additional baskets and the discrepant basket shall not be less than 1218 pounds/basket at a 95% level of confidence.

The ice condenser shall also be subdivided into 3 groups of baskets, as follows: Group 1 - Bays 1 through 8, Group 2 - Bays 9 through 16, and Group 3 - Bays 17 through 24. The minimum average ice weight of the sample baskets from Radial Rows 1, 2, 4, 6, 8, and 9 in each group shall not be less than 1218 pounds/basket at a 95% level of confidence.

The minimum total ice condenser ice weight at a 95% level of confidence shall be calculated using all ice basket weights determined during this weighing program and shall not be less than 2,368,652 pounds; and

3) Verifying, by a visual inspection of at least two flow passages per ice condenser bay, that the accumulation of frost or ice on flow passages between ice baskets, past lattice frames, through the top deck floor grating, or past the lower inlet plenum support structures and turning vanes is restricted to a thickness of less than or equal to 0.38 inch. If one flow passage per bay is found to have an accumulation of frost or ice with a thickness of greater than or equal to 0.38 inch, a representative sample of 20 additional flow passages from the same bay shall be visually inspected. If these additional flow passages are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one restricted flow passage per bay is evidence of abnormal degradation of the ice condenser.

c. At least once per 40 months by lifting and visually inspecting the accessible portions of at least two ice baskets from each one-third of the ice condenser and verifying that the ice baskets are free of detrimental structural wear, cracks, corrosion or other damage. The ice baskets shall be raised at least 12 feet for this inspection.

ICE BED TEMPERATURE MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.2 The Ice Bed Temperature Monitoring System shall be OPERABLE with at least two OPERABLE RTD channels in the ice bed at each of three basic elevations (< 11', 30'9" and 55' above the floor of the ice condenser) for each one-third of the ice condenser.

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

ACTION:

a. With the Ice Bed Temperature Monitoring System inoperable, POWER OPERATION may continue for up to 30 days provided:

   1. The ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed;

   2. The last recorded mean ice bed temperature was less than or equal to 20°F and steady or decreasing ; and

   3. The ice condenser cooling system is OPERABLE with at least:

      a) Twenty-one OPERABLE air handling units,

      b) Two OPERABLE glycol circulating pumps, and

      c) Three OPERABLE refrigerant units.

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

b. With the Ice Bed Temperature Monitoring System inoperable and with the Ice Condenser Cooling System not satisfying the minimum components OPERABILITY requirements of ACTION a.3 above, POWER OPERATION may continue for up to 6 days provided the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed and the last recorded mean ice bed temperature was less than or equal to 15°F and steady; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 The Ice Bed Temperature Monitoring System shall be determined OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours.

ICE CONDENSER DOORS

LIMITING CONDITION FOR OPERATION

3.6.5.3 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be closed and OPERABLE.

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

ACTION:

a. With one or more ice condenser doors open or otherwise inoperable (but capable of opening automatically), POWER OPERATION may continue for up to 14 days provided the ice bed temperature is monitored at least once per 4 hours and the maximum ice bed temperature is maintained less than or equal to 27°F; otherwise, restore the doors to their closed positions or OPERABLE status (as applicable) within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more ice condenser doors inoperable (not capable of opening automatically), restore all doors to OPERABLE status within 1 hour or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Inlet Doors - Ice condenser inlet doors shall be:

a. Continuously monitored and determined closed by the Inlet Door Position Monitoring System, and

b. Demonstrated OPERABLE during shutdown at least once per 3 months during the first year after the ice bed is initially fully-loaded and at least once per 6 months thereafter by:

   1) Verifying that the torque required to initially open each door is less than or equal to 675 inch pounds;

   2) Verifying that each door is capable of opening automatically and that it is not impaired by ice, frost, debris, or other obstruction;

   3) Testing a sample of at least 25% of the doors and verifying that the torque required to open each door is less than 195 inch-pounds when the door is 40 degrees open. This torque is defined as the "door opening torque" and is equal to the nominal door torque plus a frictional torque component. The doors selected for determination of the "door opening torque" shall be selected to ensure that all doors are tested at least once during four test intervals;

SURVEILLANCE REQUIREMENTS (Continued)

4) Testing a sample of at least 25% of the doors and verifying that the torque required to keep each door from closing is greater than 78 inch-pounds when the door is 40 degrees open. This torque is defined as the "door closing torque" and is equal to the nominal door torque minus a frictional torque component. The doors selected for determination of the "door closing torque" shall be selected to ensure that all doors are tested at least once during four test intervals; and

5) Calculation of the frictional torque of each door tested in accordance with Specification 4.6.5.3.1b.3) and 4), above. The calculated frictional torque shall be less than or equal to 40 inch-pounds.

4.6.5.3.2 Intermediate Deck Doors - Each ice condenser intermediate deck door shall be:

a. Verified closed and free of frost accumulation by a visual inspection at least once per 7 days, and

b. Demonstrated OPERABLE at least once per 3 months during the first year after the ice bed is initially fully-loaded and at least once per 18 months thereafter by visually verifying no structural deterioration, by verifying free movement of the vent assemblies, and by ascertaining free movement when lifted with the applicable force shown below:

	Door	Lifting Force
1)	Adjacent to Crane Wall	$\leq$ 37.4 lbs,
2)	Paired w/Door Adjacent to Crane Wall	$\leq$ 33.8 lbs,
3)	Adjacent to Containment Wall	$\leq$ 31.8 lbs, and
4)	Paired w/Door Adjacent to Containment Wall	$\leq$ 31.0 lbs.

4.6.5.3.3 Top Deck Doors - Each ice condenser top deck door shall be determined closed and OPERABLE at least once per 92 days by visually verifying:

a. That the doors are in place, and

b. That no condensation, frost, or ice has formed on the doors or blankets which would restrict their lifting and opening if required.

INLET DOOR POSITION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 The Inlet Door Position Monitoring System shall be OPERABLE.

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

ACTION:

With the Inlet Door Position Monitoring System inoperable, POWER OPERATION may continue for up to 14 days, provided the Ice Bed Temperature Monitoring System is OPERABLE and the maximum ice bed temperature is less than or equal to 27°F when monitored at least once per 4 hours; otherwise, restore the Inlet Door Position Monitoring System to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 The Inlet Door Position Monitoring System shall be determined OPERABLE by:

a. Performing a CHANNEL CHECK at least once per 12 hours,

b. Performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 18 months, and

c. Verifying that the Monitoring System correctly indicates the status of each inlet door as the door is opened and reclosed during its testing per Specification 4.6.5.3.1.

DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

LIMITING CONDITION FOR OPERATION

3.6.5.5 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be OPERABLE and closed.

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

ACTION:

With a personnel access door or equipment hatch inoperable or open except for personnel transit entry, restore the door or hatch to OPERABLE status or to its closed position (as applicable) 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

4.6.5.5.1 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined closed by a visual inspection prior to increasing the Reactor Coolant System T_{avg} above 200°F and after each personnel transit entry when the Reactor Coolant System T_{avg} is above 200°F.

4.6.5.5.2 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined OPERABLE by visually inspecting the seals and sealing surfaces of these penetrations and verifying no detrimental misalignments, cracks or defects in the sealing surfaces, or apparent deterioration of the seal material:

a. Prior to final closure of the penetration each time it has been opened, and

b. At least once per 10 years for penetrations containing seals fabricated from resilient materials.

CONTAINMENT AIR RETURN AND HYDROGEN SKIMMER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent Containment Air Return and Hydrogen Skimmer Systems shall be OPERABLE.

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

ACTION:

With one Containment Air Return and Hydrogen Skimmer System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6.1 Each Containment Air Return and Hydrogen Skimmer System shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by:

a. Verifying that the air return and hydrogen skimmer fans start automatically on a Containment Pressure-High-High test signal after a 9 ± 1 minute delay and operate for at least 15 minutes;

b. Verifying that during air return fan operation with the air return fan damper closed and with the bypass dampers open, the fan motor current is less than or equal to 59 amps when the fan speed is 1187 ± 13 rpm;

c. Verifying that with the hydrogen skimmer fan operating and the motor-operated valve in its suction line closed, the fan motor current is less than or equal to 69 amps when the fan speed is 3580 ± 20 rpm;

d. Verifying that with the air return fan off, the motor-operated damper in the air return fan discharge line to the containment's lower compartment opens automatically with a 10 ± 1 second delay after a Containment Pressure-High-High test signal;

e. Verifying that with the air return fan operating, the check damper in the air return fan discharge line to the containment's lower compartment is open;

SURVEILLANCE REQUIREMENTS (Continued)

f. Verifying that the motor-operated valve in the hydrogen skimmer suction line opens automatically and the hydrogen skimmer fans receive a start permissive signal; and

g. Verifying that with the fan off, the air return fan check damper is closed.

4.6.5.6.2 At least once per 18 months, each Containment Air Return and Hydrogen Skimmer System shall be demonstrated OPERABLE by:

a. Verifying that each air return fan is deenergized or is prevented from starting by the Containment Pressure Control System when the containment internal pressure is less than or equal to 0.25 psid, relative to the outside atmosphere; and

b. Verifying that each air return fan isolation damper closes or is prevented from opening by the Containment Pressure Control System when the containment internal pressure is less than or equal to 0.25 psid and is allowed to open at greater than or equal to 0.45 psid, relative to the outside atmosphere.

FLOOR DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.7 The ice condenser floor drains shall be OPERABLE.

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

ACTION:

With an ice condenser floor drain inoperable, restore the floor drain to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.7 Each ice condenser floor drain shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

a. Verifying that the valve gate opening is not impaired by ice, frost, or debris,

b. Verifying that the valve seat is not damaged,

c. Verifying that the valve gate opens when a force of less than or equal to 66 lbs is applied, and

d. Verifying that the drain line from the ice condenser floor to the containment lower compartment is unrestricted.

REFUELING CANAL DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.8 The refueling canal drains shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.5.8 Each refueling canal drain shall be demonstrated OPERABLE:

a. Prior to increasing the Reactor Coolant System temperature above 200°F after each partial or complete filling of the canal with water by verifying that the valves in the drain line are locked open and that the drain is not obstructed by debris, and

b. At least once per 92 days by verifying, through a visual inspection, that there is no debris that could obstruct the drain.

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

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

ACTION:

With the divider barrier seal inoperable. restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:

a. Removing two divider barrier seal test coupons and verifying that the physical properties of the test coupons are within the acceptable range of values shown in Table 3.6-3, and

b. Visually inspecting at least 95% of the seal's entire length and:

 1) Verifying that the seal and seal mounting bolts are properly installed, and

 2) Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

TABLE 3.6-3

DIVIDER BARRIER SEAL ACCEPTABLE PHYSICAL PROPERTIES

MATERIAL	TENSILE STRENGTH
Membrane Type Seals	
Mk 10	39.7 lbs
Mk 11	39.7 lbs

CONTAINMENT VALVE INJECTION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6 Both trains of the Containment Valve Injection Water System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one train of the Containment Valve Injection Water System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each train of the Containment Valve Injection Water System shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to 1.10 P_a (16.2 psig) and has adequate capacity to maintain system pressure for at least 30 days.

4.6.6.2 Each train of the Containment Valve Injection Water System shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve seal injection flow rate is less than 1.7 gpm (Unit 1), 1.6 gpm (Unit 2) for Train A and 1.4 gpm (Unit 1), 1.5 gpm (Unit 2) for Train B with a tank pressure greater than or equal to 45 psig and each automatic valve in the flow path actuates to its correct position on a Containment Pressure-High or a Containment Pressure-High-High test signal.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 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 FOUR LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	43

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					LIFT SETTING (± 1%)*	ORIFICE SIZE
	Loop A	Loop B	Loop C	Loop D		
1.	SV-20	SV-14	SV-8	SV-2	1175 psig	14.18 in.^{2}
2.	SV-21	SV-15	SV-9	SV-3	1190 psig	14.18 in.^{2}
3.	SV-22	SV-16	SV-10	SV-4	1205 psig	14.18 in.^{2}
4.	SV-23	SV-17	SV-11	SV-5	1220 psig	14.18 in.^{2}
5.	SV-24	SV-18	SV-12	SV-6	1230 psig	14.18 in.^{2}

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

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Two motor-driven 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 auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying that each motor-driven pump develops a total dynamic head of greater than or equal to 3470 feet at a flow of greater than or equal to 400 gpm;

2) Verifying that the steam turbine-driven pump develops a total dynamic head of greater than or equal to 3550 feet at a flow of greater than or equal to 400 gpm when the secondary steam supply pressure is greater than 600 psig and the auxiliary feedwater pump turbine is operating at 3600 rpm. The provisions of Specification 4.0.4 are not applicable for 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;

4) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

5) Verifying that the isolation valves in the auxiliary feedwater pump suction lines are open and that power is removed from the valve operators on Valves CA-2, CA-7A, CA-9B, and CA-11A and that the respective circuit breakers are padlocked.

b. At least once per 18 months during shutdown 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,

2) Verifying that each motor-driven auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal,

3) Verifying that the turbine-driven auxiliary feedwater pump steam supply valves open upon receipt of an Auxiliary Feedwater Actuation test signal, and

4) Verifying that the valve in the suction line of each auxiliary feedwater pump from the Nuclear Service Water System automatically actuates to its full open position within less than or equal to 15 seconds* on a Loss-of-Suction test signal.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying normal flow to each steam generator.

*Includes 5 second time delay.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.3 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

4.7.1.3 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 least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than 10 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.4 Each main steam line isolation valve (MSLIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSLIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSLIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. The provisions of Specification 3.0.4 are not applicable. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 Each MSLIV 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.

CONDENSATE STORAGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.5 The Condensate Storage System (CSS) (CA Condensate Storage Tank, Upper Surge and Condenser Hotwell) shall be OPERABLE with a contained water volume of at least 225,000 gallons of water.

APPLICABILITY: MODES 1, 2, and 3. (Unit 2)

ACTION:

With the CSS inoperable, within 4 hours either:

a. Restore the CSS to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the standby nuclear service water pond as a backup supply to the auxiliary feedwater pumps and restore the CSS to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 The CSS shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection, Phase "A" Isolation, or Phase "B" Isolation test signal, and

2) Each Component Cooling Water System pump starts automatically on a Safety Injection test signal.

3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent nuclear service water loops shall be OPERABLE.

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

ACTION:

With only one nuclear service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two nuclear service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection, or Phase "B" Isolation test signal, and

   2) Each Nuclear Service Water System pump starts automatically on a Safety Injection, or Loss-of-Offsite Power test signal.

3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND

LIMITING CONDITION FOR OPERATION

3.7.5 The standby nuclear service water pond (SNSWP) shall be OPERABLE with:

a. A minimum water level at or above elevation 570 feet Mean Sea Level, USGS datum, and

b. An average water temperature of less than or equal to 86.5°F at elevation 540 feet in the SNSWP intake structure.

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

ACTION: (Units 1 and 2)

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

SURVEILLANCE REQUIREMENTS

4.7.5 The SNSWP shall be determined OPERABLE:

a. At least once per 24 hours by verifying the water level to be within its limit,

b. At least once per 24 hours during the months of July, August, and September by verifying the water temperature to be within its limit, and

c. At least once per 12 months by visually inspecting the SNSWP dam and verifying no abnormal degradation, erosion, or excessive seepage.

3/4.7.6 CONTROL ROOM AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Room Area Ventilation Systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION: (Units 1 and 2)

MODES 1, 2, 3 and 4:

With one Control Room Area Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one Control Room Area Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Area Ventilation System with flow through the HEPA filters and carbon adsorbers.

b. With both Control Room Area Ventilation Systems inoperable, or with the OPERABLE Control Room Area Ventilation System, required to be operating by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Room Area Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 90°F;

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and carbon adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or carbon adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d* of Regulatory Guide 1.52, Revisions 2, March 1978, and the sytem flow rate is 6000 cfm ± 10%;

   2) Ver'fying, 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

   3) Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of carbon 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

e. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters, carbon adsorber banks, and moisture separators is less than 8 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10%;

   2) Verifying that on a High Radition-Air Intake, or Smoke Density-High test signal, the system automatically isolates the affected intake from outside air with recirculating flow through the HEPA filters and carbon adsorber banks;

   3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge relative to adjacent areas at less than or equal to pressurization flow of 4000 cfm to the control room during system operation;

   4) Verifying that the heaters dissipate 25 ± 2.5 kW, and

*The requirement for reducing refrigerant concentration to 0.01 ppm may be satisfied by operating the system for 10 hours with heaters on and operating.

SURVEILLANCE REQUIREMENTS (Continued)

5) Verifying that on a High Chlorine/Toxic Gas test signal, the system automatically isolates the affected intake from outside air with recirculating flow through the HEPA filters and carbon adsorbers banks within 10 seconds (plus air travel time between the detectors and the isolation dampers).

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 6000 cfm ± 10%; and

g. After each complete or partial replacement of a carbon adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) 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%.

3/4.7.7 AUXILIARY BUILDING FILTERED EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 The Auxiliary Building Filtered Exhaust System shall be OPERABLE.

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

ACTION:

With the Auxiliary Building Filtered Exhaust System inoperable, restore the inoperable system 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

4.7.7 The Auxiliary Building Filtered Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) 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 30,000 cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

*The requirement for reducing refrigerant concentration to 0.01 ppm may be satisfied by operating the system for 10 hours with heaters on and operating.

3) Verifying a system flow rate of 30,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters, charcoal adsorber banks, and moisture separators of less than 8 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ± 10%,

2) Verifying that the system starts on a Safety Injection test signal, and directs its exhaust flow through the HEPA filters and charcoal adsorbers,

3) Verifying that the system maintains the ECCS pump room at a negative pressure relative to adjacent areas,

4) Verifying that the filter cooling bypass valves can be manually opened, and

5) Verifying that the heaters dissipate 40 ± 4 kW.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 30,000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 30,000 cfm ± 10%.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in lieu of the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If less than two snubbers of each type are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type Found During Inspection	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

*The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

#The provisions of Specification 4.0.2 are not applicable.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers (regardless of type) that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.8f. When a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and may be determined OPERABLE via functional testing only if the test is started with the piston in the as-found setting extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

d. Refueling Outage Inspections

At each refueling, the systems which have the potential for a severe dynamic event, specifically, the Main Steam System (upstream of the main steam isolation valves) the main steam safety and power-operated relief valves and piping, Auxiliary Feedwater System, main steam supply to the auxiliary feedwater pump turbine, and the letdown and charging portion of the CVCS System shall be inspected to determine if there has been a severe dynamic event. In the case of a severe dynamic event, mechanical snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the mechanical snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying freedom-of-motion using one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced or repaired before returning to power. The requirements of Specification 4.7.8b. are independent of the requirements of this specification.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The large-bore steam generator hydraulic snubbers shall be treated as

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests (Continued)

a separate type (population) for functional test purposes. A 10% random sample shall be tested at least once per 18 months during refueling with continued testing based on a failure evaluation. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of all snubbers shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.8f., an additional 10% of all snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested; or

2) A representative sample of all snubbers shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.8f. The cumulative number of snubbers tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

e. Functional Tests (Continued)

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from all snubbers and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

g. Functional Test Failure Analysis (Continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service performance of all snubbers shall be monitored. If a service lifetime limit is associated (established) with any snubber (or critical part) based on manufacturer's information, qualification tests, or historical service results, then the service life shall be monitored to ensure that the service life is not exceeded between surveillance inspections. Established snubber service life shall be extended or shortened based on monitored test results and failure history. The replacements (snubbers or critical parts) shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

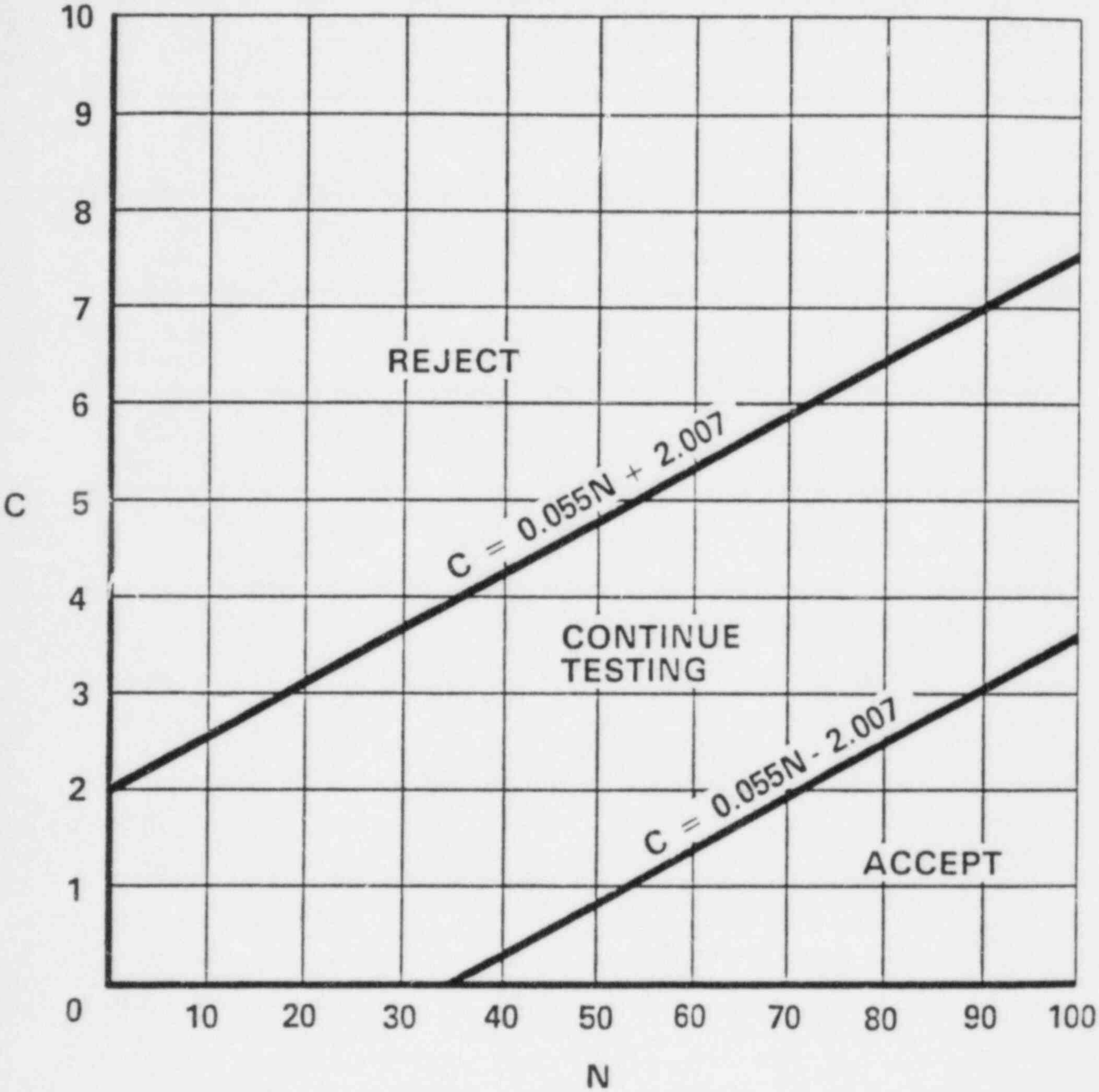

FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.3 Reports - A report shall be prepared a. ubmitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10.1 The Fire Suppression Water System shall be OPERABLE with:

a. At least two fire suppression pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header, and

b. An OPERABLE flow path capable of taking suction from Lake Wylie and transferring the water through distribution piping with OPERABLE sectionalizing control valves and isolation valves for each sprinkler, hose standpipe, or Spray System riser required to be OPERABLE per Specifications 3.7.10.2 and 3.7.10.4.

APPLICABILITY: At all times.

ACTION:

a. With one of the above required pumps and/or one Water Supply/Distribution System inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable establish a backup Fire Suppression Water System within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.10.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor-driven pump and operating it for at least 15 minutes on recirculation flow,

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path which is accessible during plant operations is in its correct position,

c. At least once per 6 months by performance of a system flush of the outside distribution loop to verify no flow blockage by fully opening the hydraulically most remote hydrant,

d. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

e. At least once per 18 months by verifying that each valve (manual, power-operated, or automatic) in the flow path which is inaccessible during plant operations is in its correct position,

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1) Verifying that each automatic valve in the flow path actuates to its correct position,

2) Verifying that each pump develops at least 2500 gpm at a net pressure of 144 psig by testing at three points on the pump performance curve,

3) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4) Verifying that each fire suppression pump starts within 10 psig of its intended starting pressure (A pump, primary switch-95 psig; B pump, primary switch-90 psig; and C pump, primary switch-85 psig).

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 8, Section 16 of the Fire Protection Handbook, 15th Edition, published by the National Fire Protection Association.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.2 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. Elevation 522 + 0 ft - Auxiliary Building

Room No.	Equipment
100	RHR & Containment Spray Sump Pump Area
101	Corridor
104	RHR Pump 1B
105	RHR Pump 1A
106	Corridor
109	RHR Pump 2B
110	RHR Pump 2A
111	Corridor
112	Corridor

b. Elevation 543 + 0 ft - Auxiliary Building

230	Cent. Chg. Pump 1A
231	Cent. Chg. Pump 1B
240	Cent. Chg. Pump 2A
241	Cent. Chg. Pump 2B
250	Unit 1 Aux. Feedwater Pump Room
260	Unit 2 Aux. Feedwater Pump Room

c. Elevation 554 + 0 ft - Auxiliary Building

340	Battery Room Corridor (DD-EE)
350	Battery Room Corridor (DD-EE)

d. Elevation 560 + 0 ft - Auxiliary Building

300	Component Cooling Pumps 1A1, 1A2, 1B1 & 1B2

e. Elevation 574 + 0 ft - Auxiliary Building

480	Cable Room Corridor (DD-EE)
490	Cable Room Corridor (DD-EE)

f. Elevation 577 + 0 ft - Auxiliary Building

400	Component Cooling Pumps 2A1, 2A2, 2B1 & 2B2

g. Reactor Buildings

-	Annulus
-	Pipe Corridor

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.10.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path which is accessible during plant operations is in its correct position,

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months by verifying that each valve (manual, power-operated, or automatic) in the flow path which is inaccessible during plant operations is in its correct position, and

d. At least once per 18 months:

1) By performing a system functional test which includes simulated automatic actuation of the system, and cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2) By a visual inspection of each Sprinkler System starting at the system isolation valve to verify the system's integrity; and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.3 The following High Pressure and Low Pressure CO_2 Systems shall be OPERABLE:

a. Low Pressure CO_2 System - Diesel generator rooms, and

b. High Pressure CO_2 System - Auxiliary feedwater pump rooms.

APPLICABILITY: Whenever equipment protected by the CO_2 Systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.10.3.1 Each of the above required CO_2 Systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

4.7.10.3.2 Each of the above required Low Pressure CO_2 Systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than 44% of full capacity, and

b. At least once per 18 months by verifying:

   1) Each system actuates manually and automatically, upon receipt of a simulated actuation signal,

   2) Damper closure devices receive an actuation signal upon system operation, and

   3) By a visual inspection of discharge nozzles to assure no blockage.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.10.3.3 Each of the above required High Pressure CO_2 Systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying the weight of each CO_2 storage cylinder to be at least 90% of full charge weight, and

b. At least once per 18 months by:

1) Verifying each system actuates manually and automatically upon receipt of a simulated actuation signal,

2) Verifying that damper closure devices receive an actuation signal upon system operation, and

3) A visual inspection of the discharge nozzles to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.10.4 The fire hose stations given in Table 3.7-3 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations given in Table 3.7-3 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to station personnel, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION requirement shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.10.4 Each of the fire hose stations given in Table 3.7-3 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station,

b. At least once per 18 months, by:

   1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2) Removing the hose for inspection and reracking, and

   3) Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

   1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

   2) Conducting a hose hydrostatic test at a pressure of 200 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-3

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK #
1. Auxiliary Building		
59, FF	522+0	1RF235
55, FF	522+0	1RF248
63-64, KK	543+0	1RF210
63, MM	543+0	1RF211
60, MM	543+0	1RF212
58, PP	543+0	1RF218
59, GG-HH	543+0	1RF236
60-61, FF-GG	543+0	1RF237
63, CC	543+0	1RF238
57, JJ	543+0	1RF242
54-55, GG	543+0	1RF249
57,FF	543+0	1RF250
52-53, GG	543+0	1RF255
51, CC	543+0	1RF256
50-51, JJ-KK	543+0	1RF262
53, MM	543+0	1RF268
50-51, NN	543+0	1RF271
62, MM-NN	560+0	1RF203
63, JJ-KK	560+0	1RF213
58, PP	560+0	1RF219
56, NN	560+0	1RF220
59, HH	560+0	1RF239
57, KK	560+0	1RF243
54-55, FF-GG	560+0	1RF251
51, KK	560+0	1RF263
52, MM-NN	560+0	1RF269
58, BB	554+0	1RF484
65, BB-CC	560+0	1RF485
62, AA-BB	560+0	1RF486
56, BB	554+0	1RF487
52, AA-BB	560+0	1RF488
49, BB-CC	560+0	1RF489
68-69, BB	560+0	1RF996
45-46, BB	560+0	1RF997
63, NN	577+0	1RF204
61, LL	577+0	1RF214
63, KK-LL	577+0	1RF215
58, PP	577+0	1RF221
59, JJ	577+0	1RF230
58, GG	577+0	1RF240
56, KK	577+0	1RF244
54, GG	577+0	1RF252
52-53, KK	577+0	1RF258
51, KK	577+0	1RF264
51-52, NN	577+0	1RF272
56, PP	577+0	1RF278
68-69, BB	577+0	1RF478

TABLE 3.7-3 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK #
65, BB-CC	577+0	1RF479
59, DD	574+0	1RF480
60, AA	574+0	1RF481
49, BB-CC	577+0	1RF490
45, BB	577+0	1RF491
55, DD	574+0	1RF492
54, AA	574+0	1RF493
63, AA	577+0	1RF993
51, AA	577+0	1RF998
62, NN	594+0	1RF205
57, MM	594+0	1RF222
63, JJ	594+0	1RF231
57, HH	594+0	1RF245
57, EE	594+0	1RF253
51, JJ	594+0	1RF259
53, NN	594+0	1RF275
64, BB	594+0	1RF984
50, BB	594+0	1RF985
51, KK	605+10	1RF265
63, JJ	605+10	1RF233
63-64, MM	631+6	1RF483
50-51, MM	631+6	1RF495

2. Fuel Pools

LOCATION	ELEVATION	HOSE RACK #
65, TT-UU	605+10	1RF208
48, TT-UU	605+10	1RF276
63-64, MM	605+10	1RF482
50-51, MM	605+10	1RF822

3. Nuclear Service Water Pump Structure

LOCATION	ELEVATION	HOSE RACK #
East Section	600+0	1RF939
West Section	600+0	1RF940

3/4.7.11 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.11 All fire barrier penetrations (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the inoperable penetration and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.11.1 At least once per 18 months the above required fire barrier penetrations and sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly;

b. At least 10% of all fire dampers. If apparent changes in appearance or abnormal degradation are found, a visual inspection of an additional 10% of the dampers shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each fire damper will be inspected every 15 years; and

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the closing mechanism and latches at least once per 6 months, and by verifying:

a. The position of each interior closed fire door at least once per 24 hours,

b. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days, and

c. That each locked closed fire door is closed at least once per 7 days.

3/4.7.12 GROUNDWATER LEVEL

LIMITING CONDITION FOR OPERATION

3.7.12 The groundwater level shall be maintained at or below the top of the adjacent floor slabs of the Reactor Containment Building and the Auxiliary Building.

APPLICABILITY: At all times.

ACTION:

a. With the groundwater level above the top of the adjacent floor slab by less than or equal to 5 feet, reduce the groundwater level to or below the top of the affected adjacent floor slab within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the groundwater level above the top of the adjacent floor slab by greater than 5 feet but less than 15 feet, reduce the groundwater level to less than or equal to 5 feet above the top of the affected adjacent floor slab within 24 hours and to or below the top of the affected adjacent floor slab within 7 days of initially exceeding the above limits or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the groundwater level above the top of the adjacent floor slab by greater than or equal to 15 feet, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours. Perform an engineering evaluation to determine the effects of this higher groundwater level on the affected building(s) and submit the results of this evaluation and any corrective action determined necessary to the Commission as a Special Report pursuant to Specification 6.9.2 prior to increasing T_{avg} above 200°F.

d. Determine the rate of rise of groundwater when the level reaches the top of the floor slab. If the rate of rise of the groundwater level is greater than or equal to 0.3 foot per hour, determine the rate of rise at least once per 30 minutes. If the rate of rise exceeds 0.5 foot per hour for more than 1 hour, be in at least HOT STANDBY within 1 hour and in COLD SHUTDOWN within the following 30 hours. If the rate of rise is less than 0.5 feet per hour, comply with the requirements of ACTIONS a., b., and c. above.

SURVEILLANCE REQUIREMENTS

4.7.12 The groundwater level shall be determined at the following frequencies by monitoring the water level and by verifying the absence of alarm in the six groundwater monitor wells as shown in FSAR Figure 2.4.13-14 installed around the perimeter of the Unit 1 Reactor and Auxiliary Building:

a. At least once per 7 days when the groundwater level is at or below the top of the adjacent floor slab, and

b. At least once per 24 hours when the groundwater level is above the top of the adjacent floor slab.

3/4.7.13 STANDBY SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 The Standby Shutdown System (SSS) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION: (Units 1 and 2)

a. With the Standby Shutdown System inoperable, restore the inoperable equipment to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With the total leakage from UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE and reactor coolant pump seal leakage greater than 26 gpm, declare the Standby Makeup Pump inoperable and take ACTION a., above.

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

SURVEILLANCE REQUIREMENTS

4.7.13.1 The Standby Shutdown System diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel level in the fuel storage tank is greater than or equal to 67 inches, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes at greater than or equal to 700 kW.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM-D975-1977 when checked for viscosity and water and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.13.2 The Standby Shutdown System diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The electrolyte level of each battery is above the plates; and

2) The overall battery voltage is greater than or equal to 24 volts.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

c. At least once per 18 months by verifying that:

1) The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, tight, and free of corrosion.

4.7.13.3 The Standby Makeup Pump water supply shall be demonstrated OPERABLE by:

a. Verifying at least once per 7 days:

1) That the requirements of Specification 3.9.10 are met and the boron concentration in the storage pool is greater than or equal to 2000 ppm, or

2) That a contained borated water volume of at least 112,320 gallons with minimum boron concentration of 2,000 ppm is available and capable of being aligned to the Standby Makeup Pump.

b. Verifying at least once per 92 days that the Standby Makeup Pump develops a flow of greater than or equal to 26 gpm at a pressure greater than or equal to 2488 psig.

4.7.13.4 The Standby Shutdown System 250/125-Volt Battery Bank and its associated charger shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) That the electrolyte level of each battery is above the plates, and

2) The total battery terminal voltage is greater than or equal to 258/129 volts on float charge.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

c. At least once per 18 months by verifying that:

1) The batteries, cell plates, and battery racks show no visual indications of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.13.5 The Steam Turbine Driven Auxiliary Feedwater Pump and associated components shall be demonstrated OPERABLE at least once per 18 months by verifying that the system functions as designed from the Standby Shutdown System.

4.7.13.6 Each Standby Shutdown System instrumentation device shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

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 Essential Auxiliary Power System, and

b. Two separate and independent diesel generators, each with:

   1) A separate day tank containing a minimum volume of 518.5 gallons of fuel,

   2) A separate Fuel Storage System containing a minimum volume of 82,056 gallons of fuel,

   3) A separate fuel transfer valve, and

   4) A separate 125 VDC battery and charger connected to the diesel generator control loads.

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

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. When in MODE 1, 2, or 3 with a steam pressure greater than 900 psig, the steam-driven auxiliary feedwater pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Specfication 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators 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. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With a diesel generator operating at greater than 5750 kW, within 1 hour reduce the diesel generator output to less than or equal to 5750 kW.

g. With the Cathodic Protection System inoperable, restore the System to OPERABLE status within 10 days or prepare and submit a Special Report pursuant to Specification 6.9.2 outlining the cause of the inoperability and the plans for restoring the System to OPERABLE.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Essential Auxiliary Power System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the day tank,

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer valve can be operated to allow fuel to be transferred from the storage system to the day tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least 441 rpm in less than or equal to 11 seconds.** The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 11 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss of offsite power by itself, or

c) Simulated loss of offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded to greater than or equal to 5600 kW but less than or equal to 5750 kW in less than or equal to 60 seconds, and operates for at least 60 minutes, and

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tank;

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

d. By verifying that the Cathodic Protection System is OPERABLE* by verifying:

1) At least once per 60 days that cathodic protection rectifiers are OPERABLE and have been inspected in accordance with the manufacturer's inspection procedures, and

2) At least once per 12 months that adequate protection from corrosion is provided in accordance with manufacturer's inspection procedures.

e. By sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

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*The Cathodic Protection System need not be OPERABLE until after June 1, 1985.

**The diesel generator start (11 sec.) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS (Continued)

1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

   a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

   b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

   c) A flash point equal to or greater than 125°F; and

   d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

f. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

g. At least once per 18 months, during shutdown, by:

   1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;

   2) Verifying the generator capability to reject a load of greater than or equal to 825 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz;

   3) Verifying the generator capability to reject a load of greater than or equal to 5600 kW but less than or equal to 5750 kW without tripping. The generator speed shall not exceed 500 rpm during and following the load rejection;

   4) Simulating a loss-of-offsite power by itself, and:

      a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be at 4160 ± 420 volts and 60 ± 1.2 Hz within 11 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

c) Verifying that all automatic diesel generator trips, except engine overspeed, low-low lube oil pressure, generator differential, and the 2 out of 3 voltage controlled overcurrent relay scheme, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) Verifying the diesel generator operates for at least 24 hours. The diesel generator shall be loaded to greater than or equal to 5600 kW but less than or equal to 5750 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 11 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits

during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2g.6)b);*

8) Verifying that the auto-connected loads to each diesel generator do not exceed 5750 kW;

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer valve transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within the tolerances given in Table 4.8-2;

13) Verifying that the voltage and diesel speed tolerances for the accelerated sequencer permissives are 92.5 ± 1% and 98 + 1%, respectively, with a minimum time delay of 2 ± 0.2 s; and

14) Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

   a) Turning gear engaged, or

   b) Maintenance mode.

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*If Specification 4.8.1.1.2g.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at greater than or equal to 5600 kW but less than or equal to 5750 kW for 1 hour or until operating temperature has stabilized.

SURVEILLANCE REQUIREMENTS (Continued)

h. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 441 rpm in less than or equal to 11 seconds; and

i. At least once per 10 years by:

   1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent, and

   2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

   3) Performing tank wall thickness measurements. The resulting data shall be evaluated and any abnormal degradation will be justified or corrected. Any abnormal degradation will be documented in a report to the Commission.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

4.8.1.1.4 Diesel Generator Batteries - Each diesel generator 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1) The electrolyte level of each battery is at or above the low mark and at or below the high mark,

   2) The overall battery voltage is greater than or equal to 125 volts on float charge, and

   3) The individual cell voltage is greater than or equal to 1.36 volts on float charge.*

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

   1) There is no visible corrosion at either terminals or connectors, and

*Two different cells shall be tested each month.

2) The average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1) The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anticorrosion material in accordance with manufacturer's recommendations,

3) The cell-to-cell pole screws torque setting is 14.5 ± 0.5 ft-lbs,

4) The battery charger will supply at least 75 amperes at a minimum of 125 volts for at least 8 hours, and

5) The battery capacity is adequate to supply and maintain in OPERABLE status its emergency loads when subjected to a battery service test. The battery shall supply a current of greater than or equal to 171.6 amps for the first minute and a current of greater than or equal to 42.5 amps for the remaining 119 minutes, while maintaining a terminal voltage of greater than or equal to 105 volts.

d. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.*

e. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

*First test to be conducted within 60 months following OL issuance date.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For purposes of this schedule, only valid tests conducted after the completion of the preoperational test requirements of Regulatory Guide 1.108, Revision 1, August 1977, shall be included in the computation of the "last 100 valid tests."

TABLE 4.8-2
LOAD SEQUENCING TIMES

LOAD GROUP NUMBER	SEQUENCE TIME (Seconds)
Initiate Timer (T_o)	9.7 ± 0.3
1 (T_1)	$T_o + 0.9 \pm 0.1$
2 (T_2)	$T_o + 1.9 \pm 0.1$
3 (T_3)	$T_o + 4.7 \pm 0.3$
4 (T_4)	$T_o + 9.4 \pm 0.6$
5 (T_5)	$T_o + 14.1 \pm 0.9$
6 (T_6)	$T_o + 18.8 \pm 1.2$
7 (T_7)	$T_o + 23.5 \pm 1.5$
8 (T_8)	$T_o + 28.2 \pm 1.8$
9 (T_9)	$T_o + 37.6 \pm 2.4$
10 (T_{10})	$T_o + 47.0 \pm 3.0$
11 (T_{11})	$T_o + 555.0 \pm 35.0$
12 (T_{12})	$T_{11} + 56.4 \pm 3.6$
13 (T_{13})	$T_{11} + 112.8 \pm 7.2$

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 Essential Auxiliary Power System, and

b. One diesel generator with:

   1) A day tank containing a minimum volume of 518.5 gallons of fuel,

   2) A fuel storage system containing a minimum volume of 82,056 gallons of fuel,

   3) A fuel transfer valve, and

   4) A 125 VDC battery and charger connected to the diesel generator control loads.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 4.5 square inch vent. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5), 4.8.1.1.3, and 4.8.1.1.4.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following D.C. channels and trains shall be OPERABLE and energized:

a. Channel 1 consisting of 125-Volt D.C. Bus No. EDA, 125-Volt D.C. Battery Bank No. EBA and a full-capacity charger,*

b. Channel 2 consisting of 125-Volt D.C. Bus No. EDB, 125-Volt D.C. Battery Bank No. EBB and a full-capacity charger,*

c. Channel 3 consisting of 125-Volt D.C. Bus No. EDC, 125-Volt D.C. Battery Bank No. EBC and a full-capacity charger,*

d. Channel 4 consisting of 125-Volt D.C. Bus No. EDD, 125-Volt D.C. Battery Bank No. EBD and a full-capacity charger,*

e. Train A consisting of 125-Volt D.C. Bus No. EDE, and

f. Train B consisting of 125-Volt D.C. Bus No. EDF.

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

ACTION:

a. With 125 VDC Bus EDE or EDF inoperable, restore the inoperable bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125 VDC Bus EDA, EDB, EDC or EDD inoperable, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With either 125 VDC Battery Bank No. EBB or EBC and/or its full-capacity charger inoperable, restore the inoperable battery and/or full-capacity charger to OPERABLE status within 10 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With either 125 VDC Battery Bank No. EBA or EBD and/or its full-capacity charger inoperable and 125 VDC diesel generator Batteries DGBA and DGBB and their full-capacity chargers in service powering Busses EDE and EDF during this period of time, restore the inoperable battery and/or full-capacity charger to OPERABLE status within

*A vital bus may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided that the vital busses associated with the other battery banks are OPERABLE and energized. Also when the spare charger is being used as a replacement for the normal battery charger verify that the A.C. input to the charger is from the same A.C. division as the normal charger which it is replacing.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

10 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two 125 VDC batteries and/or their full-capacity chargers inoperable and 125 VDC Batteries EBA and EBC and/or their full-capacity chargers in service, or 125 VDC Batteries EBB and EBD and/or their full-capacity chargers in service during this period of time, restore at least one battery and/or its full-capacity charger to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1) The parameters in Table 4.8-3 meet the Category A limits,

   2) The total battery terminal voltage is greater than or equal to 125 volts on float charge, and

   3) There is no visible indication of electrolyte leakage.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

   1) The parameters in Table 4.8-3 meet the Category B limits,

   2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohm, and

   3) The average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

   1) The cells, cell plates (if visible), and battery racks show no visual indication of physical damage or abnormal deterioration,

   2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

   3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, and

4) The battery charger will supply at least 200 amperes at a minimum of 125 volts for at least 8 hours.

d. At least once per 18 months during shutdown, by verifying that the battery capacity is adequate to either:

1) Supply and maintain in OPERABLE status all of the actual emergency loads for 1 hour when the battery is subjected to a battery service test; or

2) Supply a dummy load from Batteries EBA and EBD and from Batteries EBB and EBC of greater than or equal to 373 amperes for the first minute of the first hour, greater than or equal to 213 amperes for the next 59 minutes of the first hour and a dummy load only from Batteries EBA and EBD of greater than or equal to 210 amperes for the second hour while maintaining the battery terminal voltage greater than or equal to 105 volts

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

4.8.2.1.2 Each D.C. channel shall be determined OPERABLE and energized with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment, indicated power availability from the charger and battery, and voltage on the bus of greater than or equal to 125 volts.

TABLE 4.8-3

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts(6)	> 2.07 volts
Specific Gravity(4)	$\geq$ 1.200(5)	$\geq$ 1.195	Not more than 0.020 below the average of all connected cells or $\geq$ 1.195
		Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195(5)

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.
(4) Corrected for electrolyte temperature and level.
(5) Or battery charging current is less than 2 amps when on charge.
(6) Corrected for average electrolyte temperature.

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following D.C. electrical equipment and busses shall be OPERABLE and energized:

a. One - 125-volt D.C. train related bus (EDE or EDF),

b. Two - 125-volt D.C. channel related busses (one of which must be EDA or EDD), and

c. Two - 125-volt battery banks and full-capacity chargers associated with the above D.C. busses.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required battery banks and/or full-capacity chargers or 125-volt D.C. busses inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required battery bank and full-capacity charger to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 4.5 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.2.2.1 The above required 125-volt D.C. busses shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the bus.

4.8.2.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Specifications 4.8.2.1.1 and 4.8.2.1.2.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following A.C. electrical busses and inverters shall be OPERABLE and energized with tie breakers open between redundant busses:

a. 4160-Volt Essential Bus #ETA,

b. 4160-Volt Essential Bus #ETB,

c. 600-Volt Essential Bus #ELXA,

d. 600-Volt Essential Bus #ELXB,

e. 600-Volt Essential Bus #ELXC,

f. 600-Volt Essential Bus #ELXD,

g. 120-Volt A.C. Vital Bus # ERPA energized from Inverter # EIA connected to D.C. Channel 1,*

h. 120-Volt A.C. Vital Bus # ERPB energized from Inverter # EIB connected to D.C Channel 2,*

i. 120-Volt A.C. Vital Bus # ERPC energized from Inverter # EIC connected to D.C. Channel 3,*

j. 120-Volt A.C. Vital Bus # ERPD energized from Inverter # EID connected to D.C. Channel 4.*

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

ACTION:

a. With less than the above complement of A.C. busses OPERABLE and energized, restore the inoperable busses to OPERABLE and energized status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one inverter inoperable, energize the associated A.C. vital bus within 2 hours; restore the inoperable inverter to OPERABLE and energized 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

4.8.3.1 The specified A.C. busses and inverters shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

*An inverter may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided: (1) its vital bus is OPERABLE and energized, and (2) the vital busses associated with the other battery banks are OPERABLE and energized from their associated inverters and connected to their associated D.C. bus.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following A.C. electrical busses and inverters shall be OPERABLE and energized:

a. One - 4160-volt essential bus,

b. Two - 600-volt essential busses in a single train, and

c. Two - 120-volt A.C. vital busses energized from their respective inverters connected to their respective D.C. channels.

APPLICABILITY MODES 5 and 6.

ACTION:

With any of the above required A.C. busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required A.C. busses in the specified manner as soon as possible, and within 8 hours depressurize and vent the Reactor Coolant System through at least a 4.5 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified A.C. busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4 All containment penetration conductor overcurrent protective devices given in Tables 3.8-1a and 3.8-1b shall be OPERABLE.

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

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) given in Tables 3.8-1a and 3.8-1b inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4 All containment penetration conductor overcurrent protective devices given in Tables 3.8-1a and 3.8-1b shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1) By verifying that the medium voltage (4-15 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

      a) A CHANNEL CALIBRATION of the associated protective relays,

      b) An integrated protective system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers function as designed, and

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal Setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of a least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested; and

3) By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8-1A

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
1. 6900 VAC Swgr	
Primary Bkr RCP1A Backup Bkr 1TA-3	Reactor Coolant Pump 1A
Primary Bkr RCP1B Backup Bkr 1TB-3	Reactor Coolant Pump 1B
Primary BKR RCP1C Backup Bkr 1TC-3	Reactor Coolant Pump 1C
Primary BKR RCP1D Backup Bkr 1TD-3	Reactor Coolant Pump 1D
2. 600 VAC MCC	
1EMXC-F01B Primary Bkr Backup Fuse	Accumulator 1C Discharge Isol Vlv 1NI76A
1EMXC-F01C Primary Bkr Backup Fuse	Check Valve Test Header Cont Isol Vlv 1NI95A
1EMXC-F02A Primary Bkr Backup Fuse	Train A Alternate Power To ND LTDN Vlv 1ND1B
1EMXC-F02B Primary Bkr Backup Fuse	Hot Leg Inj. Check Vlv Test Isol Vlv 1NI153A
1EMXC-F02C Primary Bkr Backup Fuse	Cont Isol at 134 Deg Annulus Area Vlv 1VI312A
1EMXC-F03A Primary Bkr Backup Fuse	NC Pump 1C Thermal Barrier Outlet Isol Vlv 1KC345A
1EMXC-F03B Primary Bkr Backup Fuse	N_2 to Prt Cont Isol Inside Vlv 1NC54A
1EMXC-F03C Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 1NC33A

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1EMXC-F05A	
Primary Bkr	NCDT Vent Inside Cont Isol
Backup Fuse	Vlv 1WL450A
1EMXC-F05B	
Primary Bkr	Cont Sump Pumps Discharge Inside
Backup Fuse	Cont Isol Vlv 1WL825A
1EMXC-F05C	
Primary Bkr	Vent Unit Cond Drn Tank
Backup Fuse	Outside Cont Isol Vlv 1WL867A
1EMXC-F06A	
Primary Bkr	NCDT Pumps Disch Inside Cont Isol
Backup Fuse	Vlv 1WL805A
1EMXC-F07B	
Primary Bkr	Cont H_2 Purge Outlet Cont Isol
Backup Fuse	Vlv 1VY17A
1EMXD-F01A	
Primary Bkr	ND Pump 1A Suction From NC
Backup Fuse	Loop B Vlv 1ND1B
1EMXD-F01B	
Primary Bkr	Accumulator 1B Discharge
Backup Fuse	Isol Vlv 1NI65B
1EMXD-F01C	
Primary Bkr	NI Pump A to Hot Leg Check
Backup Fuse	Vlv Test Isol Vlv 1NI122B
1EMXD-F02A	
Primary Bkr	ND Pump 1B Suction from NC
Backup Fuse	Loop C Vlv 1ND36B
1EMXD-F02B	
Primary Bkr	ND to Hot Legs Chk 1NI125, 1NI129
Backup Fuse	Test Isol Vlv 1NI154B

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1EMXD-F02C Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 1NC31B
1EMXD-F05A Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 1NC35B
1EMXD-F05B Primary Bkr Backup Fuse	Rx Bldg Drain Hdr Inside Cont Isol Vlv 1KC429B
1EMXD-F05C Primary Bkr Backup Fuse	NCDT Hx Clng Water Return Inside Isol Vlv 1KC332B
1EMXD-F06A Primary Bkr Backup Fuse	NC Pump 1B Thermal Barrier Outlet Isol Vlv 1KC364B
1EMXD-F06B Primary Bkr Backup Fuse	NC Pumps Rtn Hdr Inside Cont Isol Vlv 1KC424B
1EMXK-F01A Primary Bkr Backup Fuse	UHI Check Vlv Test Line Inside Cont Isol Vlv 1NI266A
1EMXK-F01B Primary Bkr Backup Fuse	Upper Cont Vent Units Return Cont Isol Vlv 1RN429A
1EMXK-F01C Primary Bkr Backup Fuse	Backup N_2 to PORV 1NC34A From Accum Tnk 1A Vlv 1NI438A
1EMXK-F02A Primary Bkr Backup Fuse	NC Pump 1A Thermal Barrier Outlet Isol Vlv 1KC394A
1EMXK-F02B Primary Bkr Backup Fuse	Lower Cont Vent Units Return Cont Isol Vlv 1RN484A

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1EMXK-F02C	
Primary Bkr	NV Supply to Pressurizer Vlv
Backup Fuse	1NV037A
1EMXK-F03A	
Primary Bkr	S/G C Blowdown Line Sample
Backup Fuse	Inside Cont Isol Vlv 1NM210A
1EMXK-F04A	
Primary Bkr	S/G A Upper Shell Sample Inside
Backup Fuse	Cont Isol Vlv 1NM187A
1EMXK-F04B	
Primary Bkr	S/G A Blowdown Line Sample
Backup Fuse	Inside Cont Isol Vlv 1NM190A
1EMXK-F04C	
Primary Bkr	S/G C Upper Shell Sample
Backup Fuse	Inside Cont Isol Vlv 1NM207A
1EMXK-F06A	
Primary Bkr	Hydrogen Skimmer Fan 1A
Backup Fuse	Inlet Vlv 1VX1A
1EMXK-F07C	
Primary Bkr	Electric Hydrogen Recombiner
Backup Fuse	Power Supply Panel 1A
1EMXK-F09A	
Primary Bkr	Accumulator 1A Discharge Isol
Backup Fuse	Vlv 1NI54A
1EMXK-F09B	
Primary Bkr	UHI Check Vlv Test Line Inside
Backup Fuse	Cont Isol Vlv 1NI267A
1EMXK-F09C	
Primary Bkr	NC Pump Oil Fill Header
Backup Fuse	Cont Isol Vlv 1NC196A

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1EMXK-F10A Primary Bkr Backup Fuse	Containment Air Return Damper 1ARF-D-2
1EMXK-F10B Primary Bkr Backup Fuse	VQ Fans Suction From Containment Isol Vlv 1VQ2A
1EMXK-F10C Primary Bkr Backup Fuse	Cont Air Addition Containment Isol Vlv 1VQ16A
1EMXK-F11A Primary Bkr Backup Fuse	Containment Air Return Fan Motor 1A
1EMXK-F11B Primary Bkr Backup Fuse	Hydrogen Skimmer Fan Motor 1A
1EMXL-F01B Primary Bkr Backup Fuse	Trn B Alternate Power to ND Letdn Vlv 1ND37A
1EMXL-F01C Primary Bkr Backup Fuse	NI Accum D Sample Line Inside Cont Isol Vlv 1NM81B
1EMXL-F02A Primary Bkr Backup Fuse	NC Pump 1D Thermal Barrier Outlet Isol Vlv 1KC413B
1EMXL-F02B Primary Bkr Backup Fuse	Air Handling units Glycol Return Cont Isol Vlv 1NF233B
1EMXL-F02C Primary Bkr Backup Fuse	NI Accum C Sample Line Inside Cont Isol Vlv 1NM78B
1EMXL-F03A Primary Bkr Backup Fuse	S/G D Blowdown Sample Line Inside Cont Isol Vlv 1NM220B

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1EMXL-F03B Primary Bkr Backup Fuse	NI Accum A Sample Line Inside Cont Isol Vlv 1NM72B
1EMXL-F03C Primary Bkr Backup Fuse	NI Accum B Sample Line Inside Cont Isol Vlv 1NM75B
1EMXL-F04A Primary Bkr Backup Fuse	S/G B Upper Shell Sample Inside Cont Isol Vlv 1NM197B
1EMXL-F04B Primary Bkr Backup Fuse	S/G B Blowdown Sample Line Inside Cont Isol Vlv 1NM200B
1EMXL-F04C Primary Bkr Backup Fuse	S/G D Upper Shell Sample Inside Cont Isol Vlv 1NM217B
1EMXL-F06A Primary Bkr Backup Fuse	Hydrogen Skimmer Fan 1B Inlet Vlv 1VX2B
1EMXL-F06B Primary Bkr Backup Fuse	Backup N_2 to PORV 1NC32B from Accum Tnk 1B Vlv 1NI439B
1EMXL-F07C Primary Bkr Backup Fuse	Electric Hydrogen Recombiner Power Supply Panel 1B
1EMXL-F09A Primary Bkr Backup Fuse	Accumulator 1D Discharge Isol Vlv 1NI88B
1EMXL-F10A Primary Bkr Backup Fuse	Containment Air Return Damper 1ARF-D-4

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1EMXL-F10B Primary Bkr Backup Fuse	Reactor Vessel Head Vent Vlv 1NC251B
1EMXL-F10C Primary Bkr Backup Fuse	Reactor Vessel Head Vent Vlv 1NC252B
1EMXL-F11A Primary Bkr Backup Fuse	Containment Air Return Fan Motor 1B
1EMXL-F11B Primary Bkr Backup Fuse	Hydrogen Skimmer Fan Motor 1B
1EMXS-F01B Primary Bkr Backup Fuse	NC Pumps Seal Rtn Inside Cont Isol Vlv 1NV89A
1EMXS-F02A Primary Bkr Backup Fuse	ND Pump 1B Suction from NC Loop C Vlv 1ND37A
1EMXS-F02B Primary Bkr Backup Fuse	Reactor Vessel Head Vent Vlv 1NC250A
1EMXS-F03C Primary Bkr Backup Fuse	ND Pump 1A Suction from NC Loop B Vlv 1ND2A
1EMXS-F03D Primary Bkr Backup Fuse	Reactor Vessel Head Vent Vlv 1NC253A
1EMXS-F04B Primary Bkr Backup Fuse	S/G 1D Blowdown Inside Cont Isol Vlv 1BB8A
1EMXS-F04C Primary Bkr Backup Fuse	S/G 1B Blowdown Inside Cont Isol Vlv 1BB19A

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1EMXS-F05A Primary Bkr Backup Fuse	S/G 1A Blowdown Inside Cont Isol Vlv 1BB56A
1EMXS-F05B Primary Bkr Backup Fuse	S/G 1C Blowdown Inside Cont Isol Vlv 1BB60A
1EMXS-F05C Primary Bkr Backup Fuse	Pzr Liquid Sample Line Inside Cont Isol Vlv 1NM3A
1EMXS-F06A Primary Bkr Backup Fuse	Pzr Steam Sample Line Inside Cont Isol Vlv 1NM6A
1EMXS-F06B Primary Bkr Backup Fuse	NC Hot Leg A Sample Line Inside Cont Isol Vlv 1NM22A
1EMXS-F06C Primary Bkr Backup Fuse	NC Hot Leg C Sample Line Inside Cont Isol Vlv 1NM25A
1MXM-F01A Primary Bkr Backup Fuse	Reactor Coolant Pump Motor Drain Tank Pump Motor
1MXM-F02A Primary Bkr Backup Fuse	NC Pump 1B Oil Lift Pump Motor 1
1MXM-F02B Primary Bkr Backup Fuse	NC Pump 1C Oil Lift Pump Motor 1
1MXM-F03A Primary Bkr Backup Fuse	Ice Condenser Power Transformer ICT1A
1MXM-F03B Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1B6 Fan Motor A & B

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXM-F03C	
Primary Bkr	Ice Condenser Equipment Access
Backup Fuse	Door Hoist Motor 1A
1MXM-F04D	
Primary Bkr	Lighting Transformer 1LR10
Backup Fuse	
1MXM-F04E	
Primary Bkr	Lighting Transformer 1LR13
Backup Fuse	
1MXM-F05A	
Primary Bkr	175 Ton Polar Crane and 25
Backup Fuse	Ton Aux Crane No. R013 and R015
1MXM-F05C	
Primary Bkr	Upper Containment Welding Feeder
Backup Fuse	
1MXM-F06A	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A7 Fan Motor A & B
1MXM-F06B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B8 Fan Motor A & B
1MXM-F06C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A9 Fan Motor A & B
1MXM-F06D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B10 Fan Motor A & B
1MXM-F07B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A13 Fan Motor A & B
1MXM-F07C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B14 Fan Motor A & B

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXM-F08D	
Primary Bkr	Ice Condenser Refrigeration
Backup Fuse	Floor Cool Defrost Heater 1A
1MXM-F09A	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A1 Fan Motor A & B
1MXM-F09B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B2 Fan Motor A & B
1MXM-F09C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A3 Fan Motor A & B
1MXM-F09D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B4 Fan Motor A & B
1MXM-F10A	
Primary Bkr	Containment Floor and Equipment
Backup Fuse	Sump Pump Motor 1A1
1MXM-F10B	
Primary Bkr	Containment Floor and Equipment
Backup Fuse	Sump Pump Motor 1B1
1MXN-F01F	
Primary Bkr	Stud Tensioner
Backup Fuse	Hoist 1B
1MXN-F02A	
Primary Bkr	NC Pump 1B Oil Lift Pump Motor 2
Backup Fuse	
1MXN-F02B	
Primary Bkr	NC Pump 1C Oil Lift Pump Motor 2
Backup Fuse	
1MXN-F02E	
Primary Bkr	Stud Tensioner Hoist 1C
Backup Fuse	

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXN-F03A	
Primary Bkr Backup Fuse	Ice Condenser Power Transformer ICT1B
1MXN-F03B	
Primary Bkr Backup Fuse	Ice Condenser Bridge Crane 1 Crane No. R011
1MXN-F03E	
Primary Bkr Backup Fuse	Stud Tensioner Hoist 1A
1MXN-F04D	
Primary Bkr Backup Fuse	Lighting Transformer 1LR5
1MXN-F04E	
Primary Bkr Backup Fuse	Lighting Transformer 1LR6
1MXN-F05A	
Primary Bkr Backup Fuse	Ice Condenser Refrigeration Floor Cool Defrost Heater 1B
1MXN-F05B	
Primary Bkr Backup Fuse	Ice Condenser Refrigeration Floor Cool Pump Motor 1B
1MXN-F05C	
Primary Bkr Backup Fuse	Ice Condenser Equipment Access Door Hoist Motor 1B
1MXN-F06A	
Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1B1 Fan Motor A & B
1MXN-F06B	
Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1A2 Fan Motor A & B
1MXN-F06C	
Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1B3 Fan Motor A & B

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
1. 600 VAC MCC (Continued)	
1MXN-F06D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A4 Fan Motor A & B
1MXN-F07B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B5 Fan Motor A & B
1MXN-F07C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A6 Fan Motor A & B
1MXN-F08A	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B7 Fan Motor A & B
1MXN-F08B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A8 Fan Motor A & B
1MXN-F08C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B9 Fan Motor A & B
1MXN-F08D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A10 Fan Motor A & B
1MXN-F09A	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B11 Fan Motor A & B
1MXN-F09B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A12 Fan Motor A & B
1MXN-F09C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B13 Fan Motor A & B
1MXN-F09D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A14 Fan Motor A & B

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXN-F10A	
Primary Bkr Backup Fuse	Containment Floor and Equipment Sump Pump Motor 1A2
1MXN-F10B	
Primary Bkr Backup Fuse	Containment Floor and Equipment Sump Pump Motor 1B2
1MXN-F10C	
Primary Bkr Backup Fuse	Incore Instrumentation Sump Pump Motor 1
1MXN-F10D	
Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1B15 Fan Motor A & B
1MXO-F01A	
Primary Bkr Backup Fuse	Upper Containment Air Return Fan Motor 1C
1MXO-F01B	
Primary Bkr Backup Fuse	Incore Instrument Tunnel Booster Fan Motor 1A
1MXO-F02B	
Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1A
1MXO-F03A	
Primary Bkr Backup Fuse	Lower Containment Ventilation Unit 1C Fan Motor
1MXO-F04C	
Primary Bkr Backup Fuse	Upper Containment Ventilation Unit 1C Fan Motor
1MXO-F05C	
Primary Bkr Backup Fuse	Containment Pipe Tunnel Booster Fan Motor 1A
1MXP-F01A	
Primary Bkr Backup Fuse	Upper Containment Return Air Fan 1B

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXP-F01B Primary Bkr Backup Fuse	Incore Instrument Tunnel Booster Fan Motor 1B
1MXP-F02B Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1B
1MXP-F03A Primary Bkr Backup Fuse	Lower Containment Ventilation Unit 1B Fan Motor
1MXP-F04C Primary Bkr Backup Fuse	Upper Containment Ventilation Unit 1B Fan Motor
1MXP-F05C Primary Bkr Backup Fuse	Containment Pipe Tunnel Booster Fan Motor 1B
1MXQ-F01A Primary Bkr Backup Fuse	Upper Containment Return Air Fan Motor 1A
1MXQ-F01B Primary Bkr Backup Fuse	Incore Instrument Room Venti- lation Unit 1A Fan Motor
1MXQ-F02B Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1C
1MXQ-F03A Primary Bkr Backup Fuse	Lower Containment Ventilation Unit 1A Fan Motor
1MXQ-F04C Primary Bkr Backup Fuse	Upper Containment Ventilation Unit 1A Fan Motor
1MXR-F01A Primary Bkr Backup Fuse	Upper Containment Return Air Fan Motor 1D

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXR-F01B Primary Bkr Backup Fuse	Incore Instrument Room Ventilation Unit 1B Fan Motor
1MXR-F02B Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1D
1MXR-F03A Primary Bkr Backup Fuse	Lower Containment Ventilation Unit 1D Fan Motor
1MXR-F04C Primary Bkr Backup Fuse	Upper Containment Ventilation Unit 1D Fan Motor
1MXY-F02A Primary Bkr Backup Fuse	NC Pump 1A Oil Lift Pump Motor 1
1MXY-F02B Primary Bkr Backup Fuse	NC Pump 1D Oil Lift Pump Motor 1
1MXY-F02C Primary Bkr Backup Fuse	Reactor Building Lower Containment Welding Machine Receptacle 1RCPL0185
1MXY-F03A Primary Bkr Backup Fuse	Reactor Coolant Drain Tank Pump Motor 1A
1MXY-F03D Primary Bkr Backup Fuse	Ice Condenser Refrigeration Floor Cool Pump Motor 1A
1MXY-F05A Primary Bkr Backup Fuse	Lighting Transformer 1LR8
1MXY-F05B Primary Bkr Backup Fuse	Lighting Transformer 1LR11

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXY-F05C	
Primary Bkr	Lighting Transformer
Backup Fuse	1LR14
1MXY-F06A	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A5 Fan Motor A & B
1MXY-F06B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A11 Fan Motor A & B
1MXY-F06C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1B12 Fan Motor A & B
1MXY-F06D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 1A15 Fan Motor A & B
1MXY-F08A	
Primary Bkr	Incore Drive Assembly
Backup Fuse	Motor 1A
1MXY-F08B	
Primary Bkr	Incore Drive Assembly
Backup Fuse	Motor 1C
1MXY-F08C	
Primary Bkr	Incore Drive Assembly
Backup Fuse	Motor 1E
1MXY-F08D	
Primary Bkr	Lower Containment Auxiliary
Backup Fuse	Charcoal Filter Unit Fan Motor 1A
1MXZ-F02A	
Primary Bkr	NC Pump 1A Oil Lift Pump
Backup Fuse	Motor 2
1MXZ-F02B	
Primary Bkr	NC Pump 1D Oil Lift Pump
Backup Fuse	Motor 2

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXZ-F03A Primary Bkr Backup Fuse	Reactor Coolant Drain Tank Pump Motor 1B
1MXZ-F04B Primary Bkr Backup Fuse	Lighting Transformer 1LR1
1MXZ-F04C Primary Bkr Backup Fuse	Lighting Transformer 1LR2
1MXZ-F04D Primary Bkr Backup Fuse	Lighting Transformer 1LR3
1MXZ-F05A Primary Bkr Backup Fuse	Reactor Coolant Pump Jib Hoist No. R019 TH R022
1MXZ-F05C Primary Bkr Backup Fuse	Lower Containment Auxiliary Charcoal Filter Unit Fan Motor 1B
1MXZ-F06A Primary Bkr Backup Fuse	Incore Drive Assembly Motor 1B
1MXZ-F06B Primary Bkr Backup Fuse	Incore Drive Assembly Motor 1D
1MXZ-F06C Primary Bkr Backup Fuse	Incore Drive Assembly Motor 1F
1MXZ-F07B Primary Bkr Backup Fuse	Lighting Transformer 1LR4
1MXZ-F07C Primary Bkr Backup Fuse	5 Ton Jib Crane in Containment Crane No. R005

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXZ-F07D	
Primary Bkr	Reactor Cavity Manipulator
Backup Fuse	Crane No. R007 & R027
1MXZ-F08A	
Primary Bkr	Steam Generator Drain Pump
Backup Fuse	Motor 1
1MXZ-F08C	
Primary Bkr	15 Ton Equipment Access Hatch
Backup Fuse	Hoist Crane No. R009
1MXZ-F08D	
Primary Bkr	Control Rod Drive 2 Ton Jib
Backup Fuse	Hoist Crane No. R017
1MXZ-F08E	
Primary Bkr	Reactor Side Fuel Handling
Backup Fuse	Control Console
SMXG-F01C	
Primary Bkr	Standby Makeup Pump Drain Isol
Backup Fuse	Vlv 1NV876
SMXG-F05C	
Primary Bkr	Pressurizer Heaters 28, 55 & 56
Backup Fuse	
SMXG-F06A	
Primary Bkr	Standby Makeup Pump to Seal
Backup Fuse	Water Line Isol Vlv 1NV877
3. 600 VAC Pressurizer Heater Power Panels	
PHP1A-F01A	
Primary Bkr	Pressurizer Heaters
Backup Fuse	1, 2, & 22
PHP1A-F01B	
Primary Bkr	Pressurizer Heaters
Backup Fuse	5, 6, & 27

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
3. 600 VAC Pressurizer Heater Power Panels (Continued)	
PHP1A-F01C Primary Bkr Backup Fuse	Pressurizer Heaters 9, 10 & 32
PHP1A-F02C Primary Bkr Backup Fuse	Pressurizer Heaters 11, 12 & 35
PHP1A-F02D Primary Bkr Backup Fuse	Pressurizer Heaters 13, 14 & 37
PHP1A-F02E Primary Bkr Backup Fuse	Pressurizer Heaters 17, 18 & 42
PHP1B-F01A Primary Bkr Backup Fuse	Pressurizer Heaters 21, 47 & 48
PHP1B-F01B Primary Bkr Backup Fuse	Pressurizer Heaters 26, 53 & 54
PHP1B-F01C Primary Bkr Backup Fuse	Pressurizer Heaters 31, 59 & 60
PHP1B-F02C Primary Bkr Backup Fuse	Pressurizer Heaters 36, 65 & 66
PHP1B-F02D Primary Bkr Backup Fuse	Pressurizer Heaters 41, 71 & 72
PHP1B-F02E Primary Bkr Backup Fuse	Pressurizer Heaters 46, 77 & 78
PHP1C-F01A Primary Bkr Backup Fuse	Pressurizer Heaters 7, 8 & 30

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
3. 600 VAC Pressurizer Heater Power Panels (Continued)	
PHP1C-F01B	
Primary Bkr	Pressurizer Heaters
Backup Fuse	19, 20 & 45
PHP1C-F01C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	24, 51 & 52
PHP1C-F01D	
Primary Bkr	Pressurizer Heaters
Backup Fuse	29, 57 & 58
PHP1C-F02C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	34, 63 & 64
PHP1C-F02D	
Primary Bkr	Pressurizer Heaters
Backup Fuse	39, 69 & 70
PHP1C-F02E	
Primary Bkr	Pressurizer Heaters
Backup Fuse	44, 75 & 76
PHP1D-F01A	
Primary Bkr	Pressurizer Heaters
Backup Fuse	3, 4 & 25
PHP1D-F01B	
Primary Bkr	Pressurizer Heaters
Backup Fuse	15, 16 & 40
PHP1D-F01C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	23, 49 & 50
PHP1D-F02C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	33, 61 & 62
PHP1D-F02D	
Primary Bkr	Pressurizer Heaters
Backup Fuse	38, 67 & 68

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
3. 600 VAC Pressurizer Heater Power Panels (Continued)	
PHP1D-F02E Primary Bkr Backup Fuse	Pressurizer Heaters 43, 73 & 74
4. 250 VDC Reactor Building Deadlight Panelboard	
1DLD-2 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR1, 1LR2, 1LR3, 1LR4
1DLD-3 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR13, 1LR14
1DLD-4 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR5, 1LR6
1DLD-5 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR10, 1LR11
1DLD-10 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR8
5. 120 VAC Panelboards	
1ELB-5 Primary Bkr Backup Fuse	Emergency A.C. Lighting
1ELB-7 Primary Bkr Backup Fuse	Emergency A.C. Lighting
1ELB-13 Primary Bkr Backup Fuse	Emergency A.C. Lighting
1ELB-15 Primary Bkr Backup Fuse	Emergency A.C. Lighting

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
5. 120 VAC Panelboards (Continued)	
1ELB-17 Primary Bkr Backup Fuse	Emergency A.C. Lighting
1KPM-1 Primary Bkr Backup Fuse	NC Pump Motor 1A Space Heater
1KPM-2 Primary Bkr Backup Fuse	NC Pump Motor 1C Space Heater
1KPM-7-1 Primary Bkr Backup Fuse	Lower Containment Vent Unit 1A Fan Motor Space Heater
1KPM-8-1 Primary Bkr Backup Fuse	Lower Containment Vent Unit 1C Fan Motor Space Heater
1KPM-24-1 Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1A Space Heater
1KPM-24-2 Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1B Space Heater
1KPM-24-3 Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1C Space Heater
1KPM-24-4 Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1D Space Heater
1KPM-33-3, 4, 5, 6, 7 Primary Bkr Backup Fuse	Safety Injection System Temperature Transmitters
1KPN-1 Primary Bkr Backup Fuse	NC Pump Motor 1B Space Heater

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
5. 120 VAC Panelboards (Continued)	
1KPN-2	
Primary Bkr Backup Fuse	NC Pump Motor 1D Space Heater
1KPN-7-1	
Primary Bkr Backup Fuse	Lower Containment Vent Unit 1B Fan Motor Space Heater
1KPN-8-1	
Primary Bkr Backup Fuse	Lower Containment Vent Unit 1D Fan Motor Space Heater
1KPN-11	
Primary Bkr Backup Fuse	Misc Control Power for 1ATC 24
6. DC Welding Circuits	
1EQCB0001	
Primary Bkr-AA Backup Bkr-AB	Lower Containment DC Welding Circuit
1EQCB0002	
Primary Bkr-AA Backup Bkr-AB	Upper Containment DC Welding Circuit

TABLE 3.8-1B

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
1. 6900 VAC Swgr	
Primary Bkr RCP2A Backup Bkr 2TA-3	Reactor Coolant Pump 2A
Primary Bkr RCP2B Backup Bkr 2TB-3	Reactor Coolant Pump 2B
Primary BKR RCP2C Backup Bkr 2TC-3	Reactor Coolant Pump 2C
Primary BKR RCP2D Backup Bkr 2TD-3	Reactor Coolant Pump 2D
2. 600 VAC MCC	
2EMXC-F01B Primary Bkr Backup Fuse	Accumulator 2C Discharge Isol Vlv 2NI76A
2EMXC-F01C Primary Bkr Backup Fuse	Check Valve Test Header Cont Isol Vlv 2NI95A
2EMXC-F02A Primary Bkr Backup Fuse	Train A Alternate Power To ND LTDN Vlv 2ND1B
2EMXC-F02B Primary Bkr Backup Fuse	Hot Leg Inj. Check Vlv Test Isol Vlv 2NI153A
2EMXC-F02C Primary Bkr Backup Fuse	Cont Isol at 134 Deg Annulus Area Vlv 2VI312A
2EMXC-F03A Primary Bkr Backup Fuse	NC Pump 2C Thermal Barrier Outlet Isol Vlv 2KC345A
2EMXC-F03B Primary Bkr Backup Fuse	N_2 to Prt Cont Isol Inside Vlv 2NC54A
2EMXC-F03C Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 2NC33A

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXC-F05A Primary Bkr Backup Fuse	NCDT Vent Inside Cont Isol Vlv 2WL450A
2EMXC-F05B Primary Bkr Backup Fuse	Cont Sump Pumps Discharge Inside Cont Isol Vlv 2WL825A
2EMXC-F05C Primary Bkr Backup Fuse	Vent Unit Cond Drn Tank Outside Cont Isol Vlv 2WL867A
2EMXC-F06A Primary Bkr Backup Fuse	NCDT Pumps Disch Inside Cont Isol Vlv 2WL805A
2EMXC-F07B Primary Bkr Backup Fuse	Cont H_2 Purge Outlet Cont Isol Vlv 2VY17A
2EMXD-F01A Primary Bkr Backup Fuse	ND Pump 2A Suction From NC Loop B Vlv 2ND1B
2EMXD-F01B Primary Bkr Backup Fuse	Accumulator 2B Discharge Isol Vlv 2NI65B
2EMXD-F01C Primary Bkr Backup Fuse	NI Pump A to Hot Leg Check Vlv Test Isol Vlv 2NI122B
2EMXD-F02A Primary Bkr Backup Fuse	ND Pump 2B Suction from NC Loop C Vlv 2ND36B
2EMXD-F02B Primary Bkr Backup Fuse	ND to Hot Legs Chk 2NI125, 2NI129 Test Isol Vlv 2NI154B

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXD-F02C Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 2NC31B
2EMXD-F05A Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 2NC35B
2EMXD-F05B Primary Bkr Backup Fuse	Rx Bldg Drain Hdr Inside Cont Isol Vlv 2KC429B
2EMXD-F05C Primary Bkr Backup Fuse	NCDT Hx Clng Water Return Inside Isol Vlv 2KC332B
2EMXD-F06A Primary Bkr Backup Fuse	NC Pump 2B Thermal Barrier Outlet Isol Vlv 2KC364B
2EMXD-F06B Primary Bkr Backup Fuse	NC Pumps Rtn Hdr Inside Cont Isol Vlv 2KC424B
2EMXK-F01A Primary Bkr Backup Fuse	UHI Check Vlv Test Line Inside Cont Isol Vlv 2NI266A
2EMXK-F01B Primary Bkr Backup Fuse	Upper Cont Vent Units Return Cont Isol Vlv 2RN429A
2EMXK-F01C Primary Bkr Backup Fuse	Backup N_2 to PORV 2NC34A From Accum Tnk 2A Vlv 2NI438A
2EMXK-F02A Primary Bkr Backup Fuse	NC Pump 2A Thermal Barrier Outlet Isol Vlv 2KC394A
2EMXK-F02B Primary Bkr Backup Fuse	Lower Cont Vent Units Return Cont Isol Vlv 2RN484A

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXK-F02C	
Primary Bkr	NV Supply to Pressurizer Vlv
Backup Fuse	2NV037A
2EMXK-F03A	
Primary Bkr	S/G C Blowdown Line Sample
Backup Fuse	Inside Cont Isol Vlv 2NM210A
2EMXK-F04A	
Primary Bkr	S/G A Upper Shell Sample Inside
Backup Fuse	Cont Isol Vlv 2NM187A
2EMXK-F04B	
Primary Bkr	S/G A Blowdown Line Sample
Backup Fuse	Inside Cont Isol Vlv 2NM190A
2EMXK-F04C	
Primary Bkr	S/G C Upper Shell Sample
Backup Fuse	Inside Cont Isol Vlv 2NM207A
2EMXK-F06A	
Primary Bkr	Hydrogen Skimmer Fan 2A
Backup Fuse	Inlet Vlv 2VX1A
2EMXK-F07C	
Primary Bkr	Electric Hydrogen Recombiner
Backup Fuse	Power Supply Panel 2A
2EMXK-F09A	
Primary Bkr	Accumulator 2A Discharge Isol
Backup Fuse	Vlv 2NI54A
2EMXK-F09B	
Primary Bkr	UHI Check Vlv Test Line Inside
Backup Fuse	Cont Isol Vlv 2NI267A
2EMXK-F09C	
Primary Bkr	NC Pump Oil Fill Header
Backup Fuse	Cont Isol Vlv 2NC196A

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXK-F10A	
Primary Bkr	Containment Air Return Damper
Backup Fuse	2ARF-D-2
2EMXK-F10B	
Primary Bkr	VQ Fans Suction From Containment
Backup Fuse	Isol Vlv 2VQ2A
2EMXK-F10C	
Primary Bkr	Cont Air Addition Containment
Backup Fuse	Isol Vlv 2VQ16A
2EMXK-F11A	
Primary Bkr	Containment Air Return Fan
Backup Fuse	Motor 2A
2EMXK-F11B	
Primary Bkr	Hydrogen Skimmer Fan Motor 2A
Backup Fuse	
2EMXL-F01B	
Primary Bkr	Trn B Alternate Power to ND
Backup Fuse	Letdn Vlv 2ND37A
2EMXL-F01C	
Primary Bkr	NI Accum D Sample Line Inside
Backup Fuse	Cont Isol Vlv 2NM81B
2EMXL-F02A	
Primary Bkr	NC Pump 2D Thermal Barrier
Backup Fuse	Outlet Isol Vlv 2KC413B
2EMXL-F02B	
Primary Bkr	Air Handling units Glycol Return
Backup Fuse	Cont Isol Vlv 2NF233B
2EMXL-F02C	
Primary Bkr	NI Accum C Sample Line Inside
Backup Fuse	Cont Isol Vlv 2NM78B
2EMXL-F03A	
Primary Bkr	S/G D Blowdown Sample Line Inside
Backup Fuse	Cont Isol Vlv 2NM220B

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXL-F03B Primary Bkr Backup Fuse	NI Accum A Sample Line Inside Cont Isol Vlv 2NM72B
2EMXL-F03C Primary Bkr Backup Fuse	NI Accum B Sample Line Inside Cont Isol Vlv 2NM75B
2EMXL-F04A Primary Bkr Backup Fuse	S/G B Upper Shell Sample Inside Cont Isol Vlv 2NM197B
2EMXL-F04B Primary Bkr Backup Fuse	S/G B Blowdown Sample Line Inside Cont Isol Vlv 2NM200B
2EMXL-F04C Primary Bkr Backup Fuse	S/G D Upper Shell Sample Inside Cont Isol Vlv 2NM217B
2EMXL-F06A Primary Bkr Backup Fuse	Hydrogen Skimmer Fan 2B Inlet Vlv 2VX2B
2EMXL-F06B Primary Bkr Backup Fuse	Backup N_2 to PORV 2NC32B from Accum Tnk 2B Vlv 2NI439B
2EMXL-F07C Primary Bkr Backup Fuse	Electric Hydrogen Recombiner Power Supply Panel 2B
2EMXL-F09A Primary Bkr Backup Fuse	Accumulator 2D Discharge Isol Vlv 2NI88B
2EMXL-F10A Primary Bkr Backup Fuse	Containment Air Return Damper 2ARF-D-4

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXL-F10B Primary Bkr Backup Fuse	Reactor Vessel Head Vent Vlv 2NC251B
2EMXL-F10C Primary Bkr Backup Fuse	Reactor Vessel Head Vent Vlv 2NC252B
2EMXL-F11A Primary Bkr Backup Fuse	Containment Air Return Fan Motor 2B
2EMXL-F11B Primary Bkr Backup Fuse	Hydrogen Skimmer Fan Motor 2B
2EMXS-F01B Primary Bkr Backup Fuse	NC Pumps Seal Rtn Inside Cont Isol Vlv 2NV89A
2EMXS-F02A Primary Bkr Backup Fuse	ND Pump 2B Suction from NC Loop C Vlv 2ND37A
2EMXS-F02B Primary Bkr Backup Fuse	Reactor Vessel Head Vent Vlv 2NC250A
2EMXS-F03C Primary Bkr Backup Fuse	ND Pump 2A Suction from NC Loop B Vlv 2ND2A
2EMXS-F03D Primary Bkr Backup Fuse	Reactor Vessel Head Vent Vlv 2NC253A
2EMXS-F04B Primary Bkr Backup Fuse	S/G 2D Blowdown Inside Cont Isol Vlv 2BB8A
2EMXS-F04C Primary Bkr Backup Fuse	S/G 2B Blowdown Inside Cont Isol Vlv 2BB19A

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXS-F05A	
Primary Bkr	S/G 2A Blowdown Inside Cont
Backup Fuse	Isol Vlv 2BB56A
2EMXS-F05B	
Primary Bkr	S/G 2C Blowdown Inside Cont
Backup Fuse	Isol Vlv 2BB60A
2EMXS-F05C	
Primary Bkr	Pzr Liquid Sample Line Inside
Backup Fuse	Cont Isol Vlv 2NM3A
2EMXS-F06A	
Primary Bkr	Pzr Steam Sample Line Inside
Backup Fuse	Cont Isol Vlv 2NM6A
2EMXS-F06B	
Primary Bkr	NC Hot Leg A Sample Line
Backup Fuse	Inside Cont Isol Vlv 2NM22A
2EMXS-F06C	
Primary Bkr	NC Hot Leg C Sample Line
Backup Fuse	Inside Cont Isol Vlv 2NM25A
2MXM-F01A	
Primary Bkr	Reactor Coolant Pump Motor
Backup Fuse	Drain Tank Pump Motor
2MXM-F02A	
Primary Bkr	NC Pump 2B Oil Lift
Backup Fuse	Pump Motor 1
2MXM-F02B	
Primary Bkr	NC Pump 2C Oil Lift
Backup Fuse	Pump Motor 1
2MXM-F03A	
Primary Bkr	Ice Condenser Power
Backup Fuse	Transformer ICT2A
2MXM-F03B	
Primary Bkr	Ice Condenser Air Handling Unit
Backup Fuse	2B6 Fan Motor A & B

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXM-F03C Primary Bkr Backup Fuse	Ice Condenser Equipment Access Door Hoist Motor 2A
2MXM-F04D Primary Bkr Backup Fuse	Lighting Transformer 2LR10
2MXM-F04E Primary Bkr Backup Fuse	Lighting Transformer 2LR13
2MXM-F05A Primary Bkr Backup Fuse	175 Ton Polar Crane and 25 Ton Aux Crane No. R014 and R016
2MXM-F05C Primary Bkr Backup Fuse	Upper Containment Welding Feeder
2MXM-F06A Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A7 Fan Motor A & B
2MXM-F06B Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B8 Fan Motor A & B
2MXM-F06C Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A9 Fan Motor A & B
2MXM-F06D Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B10 Fan Motor A & B
2MXM-F07B Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A13 Fan Motor A & B
2MXM-F07C Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B14 Fan Motor A & B

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXM-F08D	
Primary Bkr	Ice Condenser Refrigeration
Backup Fuse	Floor Cool Defrost Heater 2A
2MXM-F09A	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2A1 Fan Motor A & B
2MXM-F09B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2B2 Fan Motor A & B
2MXM-F09C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2A3 Fan Motor A & B
2MXM-F09D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2B4 Fan Motor A & B
2MXM-F10A	
Primary Bkr	Containment Floor and Equipment
Backup Fuse	Sump Pump Motor 2A1
2MXM-F10B	
Primary Bkr	Containment Floor and Equipment
Backup Fuse	Sump Pump Motor 2B1
2MXN-F01F	
Primary Bkr	Stud Tensioner
Backup Fuse	Hoist 2B
2MXN-F02A	
Primary Bkr	NC Pump 2B Oil Lift Pump Motor 2
Backup Fuse	
2MXN-F02B	
Primary Bkr	NC Pump 2C Oil Lift Pump Motor 2
Backup Fuse	
2MXN-F02E	
Primary Bkr	Stud Tensioner Hoist 2C
Backup Fuse	

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXN-F03A	
Primary Bkr	Ice Condenser Power Transformer
Backup Fuse	ICT2B
2MXN-F03B	
Primary Bkr	Ice Condenser Bridge Crane 2
Backup Fuse	Crane No. R012
2MXN-F03E	
Primary Bkr	Stud Tensioner Hoist 2A
Backup Fuse	
2MXN-F04D	
Primary Bkr	Lighting Transformer 2LR5
Backup Fuse	
2MXN-F04E	
Primary Bkr	Lighting Transformer 2LR6
Backup Fuse	
2MXN-F05A	
Primary Bkr	Ice Condenser Refrigeration
Backup Fuse	Floor Cool Defrost Heater 2B
2MXN-F05B	
Primary Bkr	Ice Condenser Refrigeration Floor
Backup Fuse	Cool Pump Motor 2B
2MXN-F05C	
Primary Bkr	Ice Condenser Equipment Access
Backup Fuse	Door Hoist Motor 2B
2MXN-F06A	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2B1 Fan Motor A & B
2MXN-F06B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2A2 Fan Motor A & B
2MXN-F06C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2B3 Fan Motor A & B

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
1. 600 VAC MCC (Continued)	
2MXN-F06D Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A4 Fan Motor A & B
2MXN-F07B Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B5 Fan Motor A & B
2MXN-F07C Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A6 Fan Motor A & B
2MXN-F08A Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B7 Fan Motor A & B
2MXN-F08B Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A8 Fan Motor A & B
2MXN-F08C Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B9 Fan Motor A & B
2MXN-F08D Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A10 Fan Motor A & B
2MXN-F09A Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B11 Fan Motor A & B
2MXN-F09B Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A12 Fan Motor A & B
2MXN-F09C Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B13 Fan Motor A & B
2MXN-F09D Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2A14 Fan Motor A & B

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXN-F10A	
Primary Bkr	Containment Floor and Equipment
Backup Fuse	Sump Pump Motor 2A2
2MXN-F10B	
Primary Bkr	Containment Floor and Equipment
Backup Fuse	Sump Pump Motor 2B2
2MXN-F10C	
Primary Bkr	Incore Instrumentation
Backup Fuse	Sump Pump Motor 2
2MXN-F10D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2B15 Fan Motor A & B
2MXO-F01A	
Primary Bkr	Upper Containment Air Return
Backup Fuse	Fan Motor 2C
2MXO-F02B	
Primary Bkr	Control Rod Drive Vent Fan
Backup Fuse	Motor 2A
2MXO-F03A	
Primary Bkr	Lower Containment Ventilation
Backup Fuse	Unit 2C Fan Motor
2MXO-F04C	
Primary Bkr	Upper Containment Ventilation
Backup Fuse	Unit 2C Fan Motor
2MXO-F05C	
Primary Bkr	Containment Pipe Tunnel
Backup Fuse	Booster Fan Motor 2A
2MXP-F01A	
Primary Bkr	Upper Containment Return
Backup Fuse	Air Fan 2B

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXP-F02B Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 2B
2MXP-F03A Primary Bkr Backup Fuse	Lower Containment Ventilation Unit 2B Fan Motor
2MXP-F04C Primary Bkr Backup Fuse	Upper Containment Ventilation Unit 2B Fan Motor
2MXP-F05C Primary Bkr Backup Fuse	Containment Pipe Tunnel Booster Fan Motor 2B
2MXQ-F01A Primary Bkr Backup Fuse	Upper Containment Return Air Fan Motor 2A
2MXQ-F01B Primary Bkr Backup Fuse	Incore Instrument Room Ventilation Unit 2A Fan Motor
2MXQ-F02B Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 2C
2MXQ-F03A Primary Bkr Backup Fuse	Lower Containment Ventilation Unit 2A Fan Motor
2MXQ-F04C Primary Bkr Backup Fuse	Upper Containment Ventilation Unit 2A Fan Motor
2MXR-F01A Primary Bkr Backup Fuse	Upper Containment Return Air Fan Motor 2D

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXR-F01B Primary Bkr Backup Fuse	Incore Instrument Room Ventilation Unit 2B Fan Motor
2MXR-F02B Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 2D
2MXR-F03A Primary Bkr Backup Fuse	Lower Containment Ventilation Unit 2C Fan Motor
2MXR-F04C Primary Bkr Backup Fuse	Upper Containment Ventilation Unit 2D Fan Motor
2MXY-F02A Primary Bkr Backup Fuse	NC Pump 2A Oil Lift Pump Motor 1
2MXY-F02B Primary Bkr Backup Fuse	NC Pump 2D Oil Lift Pump Motor 1
2MXY-F02C Primary Bkr Backup Fuse	Reactor Building Lower Containment Welding Machine Receptacle 2RCPL0185
2MXY-F03A Primary Bkr Backup Fuse	Reactor Coolant Drain Tank Pump Motor 2A
2MXY-F03D Primary Bkr Backup Fuse	Ice Condenser Refrigeration Floor Cool Pump Motor 2A
2MXY-F05A Primary Bkr Backup Fuse	Lighting Transformer 2LR8
2MXY-F05B Primary Bkr Backup Fuse	Lighting Transformer 2LR11

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXY-F05C	
Primary Bkr	Lighting Transformer
Backup Fuse	2LR14
2MXY-F06A	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2A5 Fan Motor A & B
2MXY-F06B	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2A11 Fan Motor A & B
2MXY-F06C	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2B12 Fan Motor A & B
2MXY-F06D	
Primary Bkr	Ice Condenser Air Handling
Backup Fuse	Unit 2A15 Fan Motor A & B
2MXY-F07C	
Primary Bkr	EXH Reactor Building Equipment
Backup Fuse	Hatch Jib Cranes R035 & R036
2MXY-F08A	
Primary Bkr	Incore Drive Assembly
Backup Fuse	Motor 2A
2MXY-F08B	
Primary Bkr	Incore Drive Assembly
Backup Fuse	Motor 2C
2MXY-F08C	
Primary Bkr	Incore Drive Assembly
Backup Fuse	Motor 2E
2MXY-F08D	
Primary Bkr	Lower Containment Auxiliary
Backup Fuse	Charcoal Filter Unit Fan Motor 2A
2MXZ-F02A	
Primary Bkr	NC Pump 2A Oil Lift Pump
Backup Fuse	Motor 2

TABLE 3.8-1BA (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXZ-F02B Primary Bkr Backup Fuse	NC Pump 2D Oil Lift Pump Motor 2
2MXZ-F03A Primary Bkr Backup Fuse	Reactor Coolant Drain Tank Pump Motor 2B
2MXZ-F04B Primary Bkr Backup Fuse	Lighting Transformer 2LR1
2MXZ-F04C Primary Bkr Backup Fuse	Lighting Transformer 2LR2
2MXZ-F04D Primary Bkr Backup Fuse	Lighting Transformer 2LR3
2MXZ-F05A Primary Bkr Backup Fuse	Reactor Coolant Pump Jib Hoist No. R023 TH R026
2MXZ-F05C Primary Bkr Backup Fuse	Lower Containment Auxiliary Charcoal Filter Unit Fan Motor 2B
2MXZ-F06A Primary Bkr Backup Fuse	Incore Drive Assembly Motor 2B
2MXZ-F06B Primary Bkr Backup Fuse	Incore Drive Assembly Motor 2D
2MXZ-F06C Primary Bkr Backup Fuse	Incore Drive Assembly Motor 2F
2MXZ-F07B Primary Bkr Backup Fuse	Lighting Transformer 2LR4

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2MXZ-F07C	
Primary Bkr	5 Ton Jib Crane in Containment
Backup Fuse	Crane No. R006
2MXZ-F07D	
Primary Bkr	Reactor Cavity Manipulator
Backup Fuse	Crane No. R008 & R028
2MXZ-F08A	
Primary Bkr	Steam Generator Drain Pump
Backup Fuse	Motor 2
2MXZ-F08C	
Primary Bkr	15 Ton Equipment Access Hatch
Backup Fuse	Hoist Crane No. R010
2MXZ-F08D	
Primary Bkr	Control Rod Drive 2 Ton Jib
Backup Fuse	Hoist Crane No. R018
2MXZ-F08E	
Primary Bkr	Reactor Side Fuel Handling
Backup Fuse	Control Console
SMXG-F06B	
Primary Bkr	Standby Makeup Pump Drain Isol
Backup Fuse	Vlv 2NV876
SMXG-R05B	
Primary Bkr	Pressurizer Heaters 28, 55 & 56
Backup Fuse	
SMXG-F06C	
Primary Bkr	Standby Makeup Pump to Seal
Backup Fuse	Water Line Isol Vlv 2NV877
3. 600 VAC Pressurizer Heater Power Panels	
PHP2A-F01A	
Primary Bkr	Pressurizer Heaters
Backup Fuse	1, 2, & 22
PHP2A-F01B	
Primary Bkr	Pressurizer Heaters
Backup Fuse	5, 6, & 27

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
3. 600 VAC Pressurizer Heater Power Panels (Continued)	
PHP2A-F01C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	9, 10 & 32
PHP2A-F02C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	11, 12 & 35
PHP2A-F02D	
Primary Bkr	Pressurizer Heaters
Backup Fuse	13, 14 & 37
PHP2A-F02E	
Primary Bkr	Pressurizer Heaters
Backup Fuse	17, 18 & 42
PHP2B-F01A	
Primary Bkr	Pressurizer Heaters
Backup Fuse	21, 47 & 48
PHP2B-F01B	
Primary Bkr	Pressurizer Heaters
Backup Fuse	26, 53 & 54
PHP2B-F01C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	31, 59 & 60
PHP2B-F02C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	36, 65 & 66
PHP2B-F02D	
Primary Bkr	Pressurizer Heaters
Backup Fuse	41, 71 & 72
PHP2B-F02E	
Primary Bkr	Pressurizer Heaters
Backup Fuse	46, 77 & 78
PHP2C-F01A	
Primary Bkr	Pressurizer Heaters
Backup Fuse	7, 8 & 30

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
3. 600 VAC Pressurizer Heater Power Panels (Continued)	
PHP2C-F01B	
Primary Bkr	Pressurizer Heaters
Backup Fuse	19, 20 & 45
PHP2C-F01C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	24, 51 & 52
PHP2C-F01D	
Primary Bkr	Pressurizer Heaters
Backup Fuse	29, 57 & 58
PHP2C-F02C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	34, 63 & 64
PHP2C-F02D	
Primary Bkr	Pressurizer Heaters
Backup Fuse	39, 69 & 70
PHP2C-F02E	
Primary Bkr	Pressurizer Heaters
Backup Fuse	44, 75 & 76
PHP2D-F01A	
Primary Bkr	Pressurizer Heaters
Backup Fuse	3, 4 & 25
PHP2D-F01B	
Primary Bkr	Pressurizer Heaters
Backup Fuse	15, 16 & 40
PHP2D-F01C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	23, 49 & 50
PHP2D-F02C	
Primary Bkr	Pressurizer Heaters
Backup Fuse	33, 61 & 62
PHP2D-F02D	
Primary Bkr	Pressurizer Heaters
Backup Fuse	38, 67 & 68

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
3. 600 VAC Pressurizer Heater Power Panels (Continued)	
PHP2D-F02E	
Primary Bkr	Pressurizer Heaters
Backup Fuse	43, 73 & 74
4. 250 VDC Reactor Building Deadlight Panelboard	
2DLD-2	
Primary Bkr	Lighting Panelboard No. 2LR1,
Backup Fuse	2LR2, 2LR3, 2LR4
2DLD-3	
Primary Bkr	Lighting Panelboard No. 2LR13,
Backup Fuse	2LR14
2DLD-4	
Primary Bkr	Lighting Panelboard No. 2LR5,
Backup Fuse	2LR6
2DLD-5	
Primary Bkr	Lighting Panelboard No. 2LR10,
Backup Fuse	2LR11
2DLD-10	
Primary Bkr	Lighting Panelboard No. 2LR8
Backup Fuse	
5. 120 VAC Panelboards	
2ELB-5	
Primary Bkr	Emergency A.C. Lighting
Backup Fuse	
2ELB-7	
Primary Bkr	Emergency A.C. Lighting
Backup Fuse	
2ELB-13	
Primary Bkr	Emergency A.C. Lighting
Backup Fuse	
2ELB-15	
Primary Bkr	Emergency A.C. Lighting
Backup Fuse	

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
5. 120 VAC Panelboards (Continued)	
2ELB-17 Primary Bkr Backup Fuse	Emergency A.C. Lighting
2KPM-1 Primary Bkr Backup Fuse	NC Pump Motor 2A Space Heater
2KPM-2 Primary Bkr Backup Fuse	NC Pump Motor 2C Space Heater
2KPM-7-1 Primary Bkr Backup Fuse	Lower Containment Vent Unit 2A Fan Motor Space Heater
2KPM-8-1 Primary Bkr Backup Fuse	Lower Containment Vent Unit 2C Fan Motor Space Heater
2KPM-24-1 Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 2A Space Heater
2KPM-24-2 Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 2B Space Heater
2KPM-24-3 Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 2C Space Heater
2KPM-24-4 Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 2D Space Heater
2KPM-33-3, 4, 5, 6, 7 Primary Bkr Backup Fuse	Safety Injection System Temperature Transmitters
2KPN-1 Primary Bkr Backup Fuse	NC Pump Motor 2B Space Heater

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
5. 120 VAC Panelboards (Continued)	
2KPN-2 Primary Bkr Backup Fuse	NC Pump Motor 2D Space Heater
2KPN-7-1 Primary Bkr Backup Fuse	Lower Containment Vent Unit 2B Fan Motor Space Heater
2KPN-8-1 Primary Bkr Backup Fuse	Lower Containment Vent Unit 2D Fan Motor Space Heater
2KPN-11 Primary Bkr Backup Fuse	Misc Control Power for 2ATC 24
6. DC Welding Circuits	
2EQCB0001 Primary Bkr - AA Backup Bkr - AB	Lower Containment DC Welding Circuit
2EQCB0002 Primary Bkr - AA Backup Bkr - AB	Upper Containment DC Welding Circuit

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 2000 ppm.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 Valves NV-231, NV-237, NV-240, NV-241, and NV-244 shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

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*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 and operating with Alarm Setpoints at 0.5 decade above steady-state count rate, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

  1) Closed by an isolation valve, blind flange, or manual valve, or

  2) Exhausting through an OPERABLE Reactor Building Containment Purge System HEPA filters and charcoal adsorbers.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or exhausting through an OPERABLE Reactor Building Containment Purge System with the capability of being automatically isolated upon heater failure within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Verifying the upper and lower containment purge supply and exhaust valves close upon a High Relative Humidity test signal.

4.9.4.2 The Reactor Building Containment Purge System shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filters and carbon adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or carbon adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedures 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 25,000 cfm ± 10% (both exhaust fans operating);

   2) Verifying within 31 days after removal, that a laboratory analysis of a presentative 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 6%; and

   3) Verifying a system flow rate of 25,000 cfm ± 10% (both exhaust fans operating) during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of carbon 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 6%;

d. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters, carbon adsorber banks, and prefilters is less than 8 inches Water Gauge while operating the system at a flow rate of 25,000 cfm ± 10% (both exhaust fans operating); and

*The requirement for reducing refrigerant concentration to 0.01 ppm may be satisfied by operating the system for 10 hours with heaters on and operating.

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that the filter train duct heater dissipates 120 ± 12 kW.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 25,000 cfm ± 10% (both exhaust fans operating); and

f. After each complete or partial replacement of a carbon adsorber bank, by verifying that the cleanup system bank satisifies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 25,000 cfm ± 10% (both exhaust fans operating).

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The reactor building manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

 1) A minimum capacity of 3250 pounds, and

 2) An overload cutoff limit less than or equal to 2900 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

 1) A minimum capacity of 610 pounds, and

 2) A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cutoff when the crane load exceeds 2850 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 610 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 3000 pounds* shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

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

SURVEILLANCE REQUIREMENTS

4.9.7 The weight of each load, other than a fuel assembly and control rod, shall be verified to be less than or equal to 3000 pounds prior to moving it over fuel assemblies.*

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*Weir gates of the spent fuel pool may be moved by crane over the stored fuel provided the spent fuel has decayed for at least 17.5 days since last being part of a core at power.

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 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 residual heat removal loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required residual heat removal 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 residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

*The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal loops shall be OPERABLE, and at least one residual heat removal 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 residual heat removal loops OPERABLE, immediately initiate corrective action to return the required residual heat removal loops to OPERABLE status, or establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no residual heat removal loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required residual heat removal loop to operation. 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 residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

*Prior to initial criticality, the residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.9 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.10 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the 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 Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.11 FUEL HANDLING VENTILATION EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.11 At least one train of the Fuel Handling Ventilation Exhaust System shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With both trains of the Fuel Handling Ventilation Exhaust System inoperable, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until the Fuel Handling Ventilation Exhaust System is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.11.1 One train of the Fuel Handling Ventilation Exhaust System shall be determined to be operating and discharging through the HEPA filter and carbon adsorbers at least once per 12 hours whenever irradiated fuel is being moved in the storage pool and during crane operation with loads over the storage pool.

4.9.11.2 Both trains of the Fuel Handling Ventilation Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and carbon adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or carbon adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d* of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 16,565 cfm ± 10%;

*The requirement for reducing refrigerant concentration to 0.01 ppm may be satisfied by operating the system for 10 hours with heaters on and operating.

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positions C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

3) Verifying a system flow rate of 16,565 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of carbon adsorber operation in any train 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%.

d. At least once per 18 months for each train by:

1) Verifying that the pressure drop across the combined HEPA filters, carbon adsorber banks, and moisture separators is less than 8 inches Water Gauge while operating the system at a flow rate of 16,565 cfm ± 10%.

2) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to ¼ inch Water Gauge relative to the outside amosphere during system operation,

3) Verifying that the filter cooling bypass valves can be manually opened, and

4) Verifying that the heaters dissipate 80 ± 8 kW.

e. After each complete or partial replacement of a HEPA filter bank in any train, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 16,565 cfm ± 10%; and

f. After each complete or partial replacement of a carbon adsorber bank in any train, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% (Unit 1), 0.05% (Unit 2) in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 16,565 cfm ± 10%.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours 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 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of 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 least once per hour 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 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

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*This requirement is not applicable during the initial calibration of the Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[1] (μCi/ml)
1. Batch Waste Release Tanks[2] Any tank which discharges liquid wastes by the liquid effluent monitor, EMF-49	P Each Batch	P Each Batch	Principal Gamma Emitters[3]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[4]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[4]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases[5] Conventional Waste Water Treatment Line	Continuous[6]	W Composite[6]	Principal Gamma Emitters[3]	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Continuous[6]	M Composite[6]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous[6]	Q Composite[6]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \text{ " } Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an _a_ _priori_ (before the fact) limit representing the capability of a measurement system and not as an _a_ _posteriori_ (after the fact) limit for a particular measurement.

(2)A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(3) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137 and Ce-141. The LLD for Ce-144 is 5×10^{-6} μCi/ml. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include: (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141, Safe Drinking Water Act.*

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

*The requirements of ACTION a.(1) and (2) are applicable only if drinking water supply is taken from the receiving water body within 3 miles downstream of the plant discharge.

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-3) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

    1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each temporary unprotected outdoor tank shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above tanks 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 Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

CHEMICAL TREATMENT PONDS

LIMITING CONDITION FOR OPERATION

3.11.1.5 The quantity of radioactive material contained in each chemical treatment pond shall be limited by the following expression:

$$\frac{264}{V} \cdot \sum_j \frac{A_j}{C_j} < 1.0$$

excluding tritium and dissolved or entrained noble gases,

Where:

A_j = pond inventory limit for single radionuclide "j", in Curies;

C_j = 10 CFR Part 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", microCuries/ml;

V = design volume of liquid and slurry in the pond, in gallons; and

264 = conversion unit, microCuries/Curie per milliliter/gallon.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed ponds exceeding the above limit, immediately suspend all additions of radioactive material to the pond and initiate corrective action to reduce the contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.5 The quantity of radioactive material contained in each batch of resin/water slurry to be transferred to the chemical treatment ponds shall be determined to be within the above limit by analyzing a representative sample of the batch to be transferred to the chemical treatment ponds shall be limited by the expression:

$$\sum_j \frac{c_j}{C_j} < 0.006$$

Where:

c_j = radioactive resin/water slurry concentration for radionuclide "j" entering the UNRESTRICTED AREA chemical treatment ponds, in microCuries/milliliter; and

C_j = 10 CFR Part 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", in microCuries/milliliter.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-4) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[1] (μCi/ml)
1. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[2]	1×10^{-4}
2. Containment Purge	P Each PURGE[3] Grab Sample	P Each PURGE[3]	Principal Gamma Emitters[2]	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
3. Unit Vent	W[3],[4] Grab Sample	W[3]	Principal Gamma Emitters[2]	1×10^{-4}
			H-3 (oxide)	1×10^{-6}
4. Containment Air Release and Addition System	D[3][5] Grab Sample	D[3][5]	Principal Gamma Emitters[2]	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
5. All Release Types as listed in 3. above.	Continuous[6]	D[7] Charcoal Sample	I-131	1×10^{-11}
			I-133	1×10^{-9}
	Continuous[6]	D[7] Particulate Sample	Principal Gamma Emitters[2]	1×10^{-10}
	Continuous[6]	M Composite Particulate Sample	Gross Alpha[8]	1×10^{-11}
	Continuous[6]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate. (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

TABLE NOTATIONS (Continued)

(2) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, and Ce-141 in Iodine and particulate releases. The LLD for Ce-144 is 5×10^{-9} µCi/ml. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3) Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER stabilization (power level constant at desired power level) after a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period, for at least one of the three gaseous release types with this notation.

(4) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

(5) Required sampling and analysis frequency during effluent release via this pathway.

(6) The ratio of the sample flow volume to the sampled stream flow volume shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(7) Samples shall be changed at least once per 24 hours and analyses shall be completed within 48 hours after changing, or after removal from sampler.

(8) The composite filter(s) will be analyzed for alpha activity by analyzing one filter per week to ensure that at least four filters are analyzed per collection period.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-4) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose 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 and beyond the SITE BOUNDARY (see Figure 5.1-4) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the WASTE GAS HOLDUP SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-4) would exceed either:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

4.11.2.4.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and WASTE GAS HOLDUP SYSTEM shall be considered OPERABLE by meeting Specification 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume; then take ACTION a. above.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

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 97,000 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 the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the Solid Radwaste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM;

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste; and

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations shall be made including direct radiation contributions from the units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12 1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6.

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*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation(2)	Forty routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY An outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site; and The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly.	Gamma dose quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne Radioiodine and Particulates	Samples from five locations. Three samples from close to the three SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground-level D/Q; One sample from the vicinity of a community having the highest calculated annual average ground-level D/Q; and One sample from a control location, as for example 15 to 30 km distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;(3) and gamma isotopic analysis(4) of composite (by location) quarterly.
3. Waterborne			
a. Surface(5)	One sample upstream. One sample downstream.	Composite sample over 1-month period.(6)	Gamma isotopic analysis(4) monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from one or two sources only if likely to be affected(7)	Quarterly.	Gamma isotopic(4) and tritium analysis quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (Continued)			
c. Drinking	One sample of each of one to three of the nearest water supplies that could be affected by its discharge. One sample from a control location.	Composite sample over 2-week period[6] when I-131 analysis is performed; monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year[8]. Composite for gross beta and gamma isotopic analyses[4] monthly. Composite for tritium analysis quarterly.
d. Sediment from Shoreline	One sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis[4] semiannually.
4. Ingestion			
a. Milk	Samples from milking animals in three locations within 5 km distance having the highest dose potential. If there are none, then one sample from milking animals in each of three areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr.[8] One sample from milking animals at a control location 15 to 30 km distant and in the least prevalent wind direction.	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic[4] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
b. Fish and Invertebrates	One sample each of a predatory species, a bottom feeder and a forage species in vicinity of plant discharge area.	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis(4) on edible portions.
	One sample each of a predatory species, a bottom feeder and a forage species in areas not influenced by plant discharge.		
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest(9).	Gamma isotopic analyses(4) on edible portion.
	Samples of three different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q if milk sampling is not performed.	Monthly, when available.	Gamma isotopic(4) and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 15 to 30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly, when available.	Gamma isotopic(4) and I-131 analysis.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS

(1) Specific parameters of distance and direction sector from the centerline of the station, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program. In lieu of any Licensee Event Report required by Specification 6.9.1 and pursuant to Specification 6.9.1.7, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

(2) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. (The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information within minimal fading.)

(3) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(5) The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities

(6) A composite sample is one in which the rate at which the liquid sampled is uniform and in which the method of sampling employed results in a specimen that is representative of the time averaged concentration at the location being sampled. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(7) Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

(8) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(9) If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000(1)				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

(1) For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/ℓ may be used.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[1] [2]

LOWER LIMIT OF DETECTION (LLD)[3]

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[4]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

*If no drinking water pathway exists, a value of 3000 pCi/ℓ may be used.

TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentrations of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between environmental collection, or end of the sample collection period, and time of counting (s).

Typical values of E, V, Y and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

(4) LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this Land Use Census was conducted. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s), with information supporting the change in the sampling locations.

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

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*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4.c. shall be followed, including analysis of control samples.

SURVEILLANCE REQUIREMENTS

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requirements stated in these Technical Specifications and those stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act, shall take precedence and shall be met.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires two independent ECCS subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, if both the required ECCS subsystems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Spray System is inoperable. Under the requirements of Specification 3.0.3 if both the required Containment Spray Systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours. It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that provided for operation in subsequent lower OPERATION MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperability is discovered in a mode lower than the applicable mode. For example, if the Containment Spray System was discovered to be inoperable while in STARTUP, the ACTION Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would still be available before the plant would be required to be in COLD SHUTDOWN. However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with: (1) the full complement of required systems, equipment, or components OPERABLE and (2) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when STARTUP with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.0.5 This specification delineates the applicability of each specification to Unit 1 and Unit 2 operation.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements. Items may be determined inoperable during use, during surveillance tests, or in accordance with this specification. Therefore, ACTION statements are entered when the Surveillance Requirements should have been performed rather than at the time it is discovered that the tests were not performed.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant STARTUP or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, boron concentration, and T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled Reactor Coolant System cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -3.7×10^{-4} $\Delta k/k/^{o}F$ The MTC value of -2.8×10^{-4} $\Delta k/k/^{o}F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -3.7×10^{-4} $\Delta k/k/^{o}F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the P-12 interlock is above its setpoint, (4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (5) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the coolant average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 16,321 gallons of 7000 ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000 ppm borated water from the refueling water storage tank.

With the coolant temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 285°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 906 gallons of 7000 ppm borated water from the boric acid storage tanks or 3170 gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the refueling water storage tank also ensure a pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120 and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

MOVABLE CONTROL ASSEMBLIES (Continued)

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to design limit DNBR during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least two of four or two of three OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, coolant flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

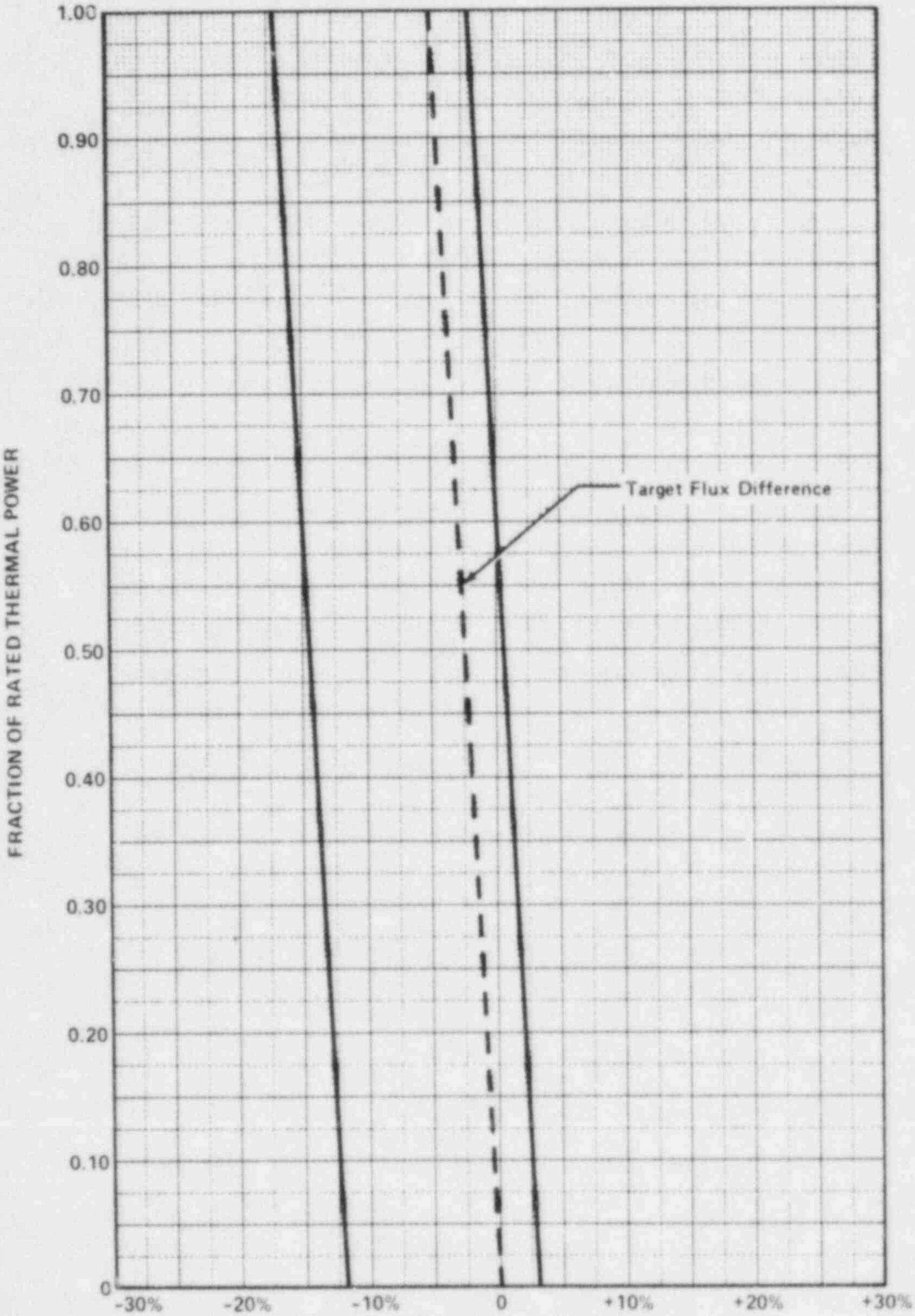


FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, Reactor Coolant System flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. The rod bow penalty as a function of burnup applied for $F^N_{\Delta H}$ is calculated with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979, and the maximum rod bow penalty is 2.7% DNBR. Since the safety analysis is performed with plant-specific safety DNBR limits of 1.49 and 1.47 compared to the design DNBR limits of 1.34 and 1.32, respectively, for the typical and thimble cells, there is a 10% thermal margin available to offset the rod bow penalty of 2.7% DNBR.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.9 was determined from expected power control manuevers over the full range of burnup conditions in the core.

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When Reactor Coolant System flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. Measurement errors of 2.1% for Reactor Coolant System total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-3. Any fouling which might bias the Reactor Coolant System flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated Reactor Coolant System flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore

QUADRANT POWER TILT RATIO (Continued)

flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. The normal locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. Alternate locations are available if any of the normal locations are unavailable.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design limit DNBR throughout each analyzed transient. The indicated T_{avg} value of 592.5°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 595°F and 2205 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. Measurement uncertainties must be accounted for during the periodic surveillance.

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and (3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. (Implementation of quarterly testing of RTS is being postponed until after approval of a similar testing interval for ESFAS). The NRC Safety Evaluation Report for WCAP-10271 was provided in a letter dated February 21, 1985 from C. O. Thomas (NRC) to J. J. Sheppard (WOG-CP&L).

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) nuclear service water pumps start and automatic valves position, and (11) component cooling pumps start and automatic valves position.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 Defeats the manual block of Safety Injection actuation on low pressurizer pressure and low steam line pressure and defeats steam line isolation on negative steam line pressure rate. Defeats the manual block of the motor-driven auxiliary feedwater pumps on trip of main feedwater pumps and low-low steam generator water level.

P-12 On decreasing reactor coolant loop temperature, P-12 automatically blocks steam dump and allows manual bypass of steam dump block for the cooldown valves only. On increasing reactor coolant loop temperature, P-12 automatically defeats the manual bypass of the steam dump block.

P-14 On increasing steam generator level, P-14 automatically trips all feedwater isolation valves, pumps and turbine and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, REMOTE SHUTDOWN SYSTEM (Continued)

control and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shutdown the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG 0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection Systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility Fire Protection Program.

Fire detectors that are used to actuate Fire Suppression Systems represent a more critically important component of a plant's Fire Protection Program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors, represents a significant degradation of fire protection for

FIRE DETECTION INSTRUMENTATION (Continued)

any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE-PART DETECTION SYSTEM

The OPERABAILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the Reactor System and avoid or mitigate damage to Reactor System components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitor used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1 x 10-6 μCi/cc are measurable.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment, or structures.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain calculated DNBR above the design DNBR value during Condition I and II events. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat; however, single failure considerations require that three loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or residual heat removal loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either residual heat removal or Reactor Coolant System) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single residual heat removal loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two residual heat removal loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one residual heat removal pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more cold legs less than or equal to 285°F are provided to prevent pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The Reactor Coolant System will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the Reactor Coolant System from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating residual heat removal loop, connected to the Reactor Coolant System, provides overpressure

SAFETY VALVES (Continued)

relief capability and will prevent overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the Reactor Coolant System from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the Reactor Coolant System is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve Reactor Coolant System pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Testing of the PORVs includes the emergency N_2 supply from the Cold Leg Accumulators. This test demonstrates that the valves in the supply line operate satisfactorily and that the nonsafety portion of the instrument air system is not necessary for proper PORV operation.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam

STEAM GENERATORS (Continued)

generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary.

These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the Reactor Coolant System, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the Reactor Coolant System ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal Reactor Coolant System pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any Reactor Coolant System pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for Reactor Coolant System pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry minimize the potential for corrosion of the Reactor Coolant System and reduce the potential for Reactor Coolant System leakage or failures due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor to secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Catawba site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microCurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6-month consecutive period with greater than 1 microCurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively, and

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the vessel are determined in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the NRC Branch Technical Position MTEB 5-2, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

PRESSURE/TEMPERATURE LIMITS (Continued)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 16 effective full power years (EFPY) of service life. The 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the pressure vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the pressure vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

TABLE B 3/4.4-1a
REACTOR VESSEL TOUGHNESS (UNIT 1)

COMPONENT	HEAT NO.	MAT'L SPEC. NO.	CU %	P %	T_{NDT} °F	50-FT-LBS 35 MIL TEMP °F	RT_{NDT} °F	SHELF ENERGY NPWD* FT-LB
Closure Head Dome	55888-1	A533B, CL.1	-	.011	-4	70	10	86
Closure Head Ring	007055	A508, CL.2	-	.006	16	23	16	101
Closure Head Flange	527038	A508, CL.2	.05	.013	-4	-2	-4	104
Vessel Flange	411212	A508, CL.2	-	.004	-31	-47	-31	153
Inlet Nozzle	526827	A508, CL.2	.05	.010	-13	7	-13	87
Inlet Nozzle	526829	A508, CL.2	.07	.010	-4	43	-4	86
Inlet Nozzle	526859	A508, CL.2	.04	.013	-4	23	-4	81
Inlet Nozzle	526857	A508, CL.2	.05	.012	-13	23	-4	77
Outlet Nozzle	526827	A508, CL.2	.05	.011	-22	27	-22	84
Outlet Nozzle	526829	A508, CL.2	.07	.010	-4	2	-4	87
Outlet Nozzle	526859	A508, CL.2	.04	.011	-13	38	-13	81
Outlet Nozzle	526857	A508, CL.2	.05	.013	-4	38	-4	60
Nozzle Shell	411077	A508, CL.2	-	.007	-40	34	-26	101
Inter. Shell	411343	A508, CL.2	.08	.004	-40	52	-8	100
Lower Shell	527708	A508, CL.2	.04	.008	-13	16	-13	101
Bottom Head Ring	527428	A508, CL.2	.06	.013	-4	74	14	68
Bottom Head Segment	55292-1	A533B, CL.1	-	.006	-22	5	-22	79
Bottom Head Segment	55292-1	A533B, CL.1	-	.006	-13	34	-13	79
Bottom Head Segment	55163-2	A533B, CL.1	-	.011	-4	38	-4	80
Bottom Head Segment	55163-2	A533B, CL.1	-	.011	-13	74	14	70
Bottom Head Dome	55178-1	A533B, CL.1	-	.010	-31	84	24	64
Nozzle Shell to Weld (P710)								
Inter Shell to Lower Shell Weld Root (P710)			.03	.009	0**	-	0**	-
Lower Shell to Bot. Head Ring Weld (P710)								
Inter Shell to Lower Shell Weld (R747)			.05	.010	-76	-9	-51	134

*Estimated per NRC Standard Review Plan Section 5.3.2 from data obtained in the principal direction.

**Estimated per NRC Standard Review Plan Section 5.3.2 from charpy tests performed at 10° F.

TABLE B 3/4.4-1b
REACTOR VESSEL TOUGHNESS (UNIT 2)

COMPONENT	HEAT NO.	MAT'L SPEC. NO.	CU %	P %	T_{NDT} °F	50-FT-LBS 35 MIL TEMP °F	RT_{NDT} °F	SHELF ENERGY NPWD* FT-LB
Closure Head Dome	B8607-1	A533B, CL.1	.13	.007	-40	50	-10	106
Closure Head Torus	B8608-1	A533B, CL.1	.07	.007	-20	57	-3	118
Closure Head Flange	B8601-1	A508, CL.2	-	.010	10	<10	10	152
Vessel Flange	B8602-1	A508, CL.2	-	.010	10	<10	10	175
Inlet Nozzle	B8609-1	A508, CL.2	-	.010	-20	<10	-20	119
Inlet Nozzle	B8609-2	A508, CL.2	-	.010	-20	<10	-20	124
Inlet Nozzle	B8609-3	A508, CL.2	-	.008	-20	<40	-20	109
Inlet Nozzle	B8609-4	A508, CL.2	-	.006	-20	97	37	97
Outlet Nozzle	B8610-1	A508, CL.2	-	.008	-10	<10	-10	141
Outlet Nozzle	B8610-2	A508, CL.2	-	.006	-10	<50	-10	144
Outlet Nozzle	B8610-3	A508, CL.2	-	.004	-20	<40	-20	140
Outlet Nozzle	B8610-4	A508, CL.2	-	.006	-10	<50	-10	150
Nozzle Shell	B8604-1	A533B, CL.1	.11	.007	-10	84	24	96
Nozzle Shell	B8604-2	A533B, CL.1	.11	.007	-10	86	26	89
Nozzle Shell	B8604-3	A533B, CL.1	.07	.009	-20	110	50	70
Inter. Shell**	B8605-1	A533B, CL.1	.09	.011	-10	75	15	89
Inter. Shell**	B8605-2	A533B, CL.1	.07	.009	-20	93	33	82
Inter. Shell**	B8616-1	A533B, CL.1	.05	.010	0	72	12	92
Lower Shell	B8806-1	A533B, CL.1	.05	.009	-60	66	6	83
Lower Shell	B8806-2	A533B, CL.1	.05	.007	-40	50	-10	102
Lower Shell	B8806-3	A533B, CL.1	.05	.006	-40	68	8	105
Bottom Head Torus	B8613-1	A533B, CL.1	.14	.010	-40	52	-8	113
Bottom Head Dome	B8612-1	A533B, CL.1	.14	.010	-40	65	5	124
Nozzle Shell Vert. Weld Seams (G1.36)			.15	.012	0*	<10	0*	>112
Nozzle Shell to Inter. Shell Weld Seam (G1.50)			.13	.016	-40	<20	-40	>102
Inter. & Lower Shell Vert. Weld Seams (G1.45)			.04	.005	-80	<-20	-80	>130
Inter. to Lower Shell Weld Seam (G1.45)								

*Estimated per NRC Standard Review Plan Section 5.3.2.
**Used for Surveillance Program Weldment.

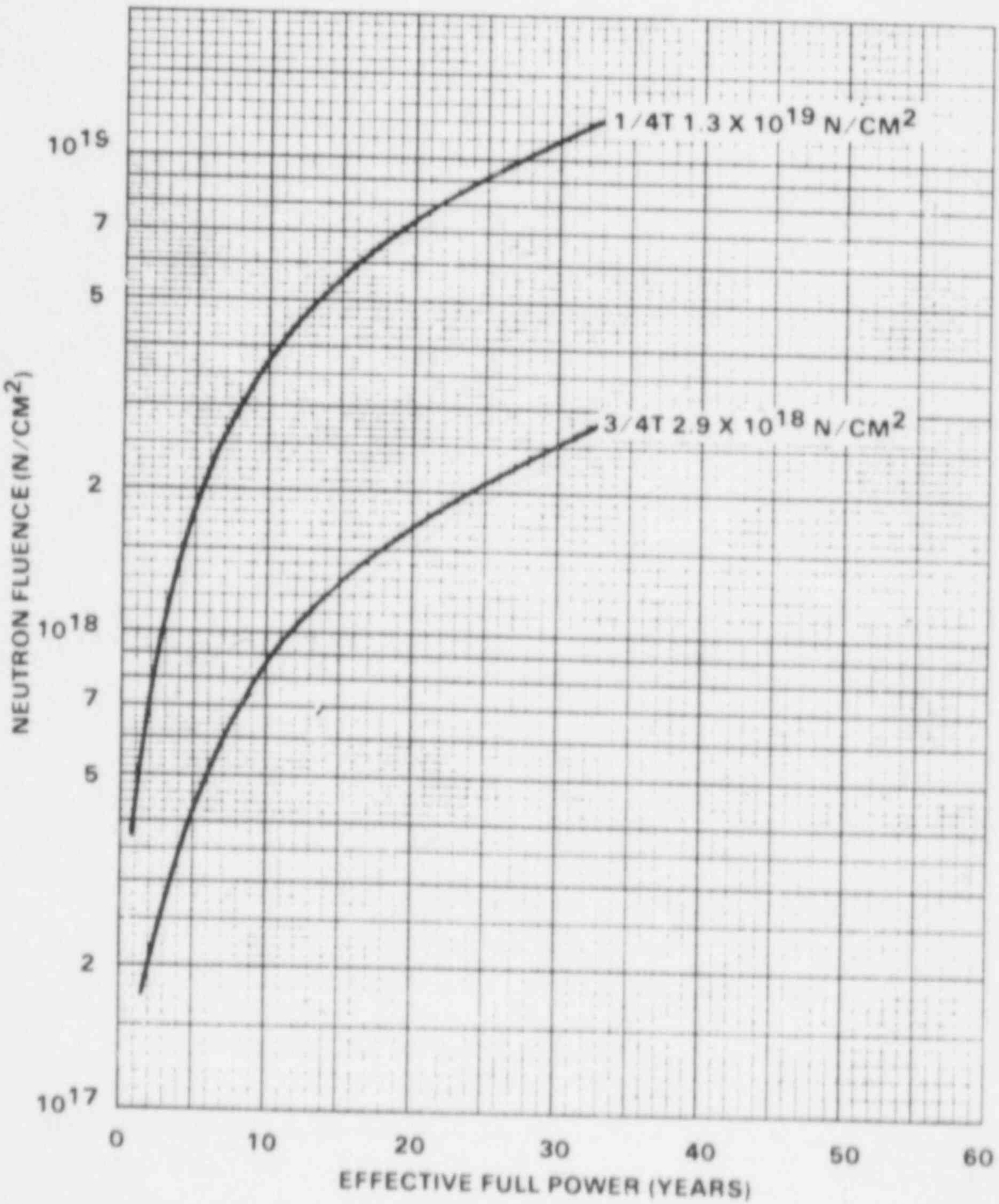


FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

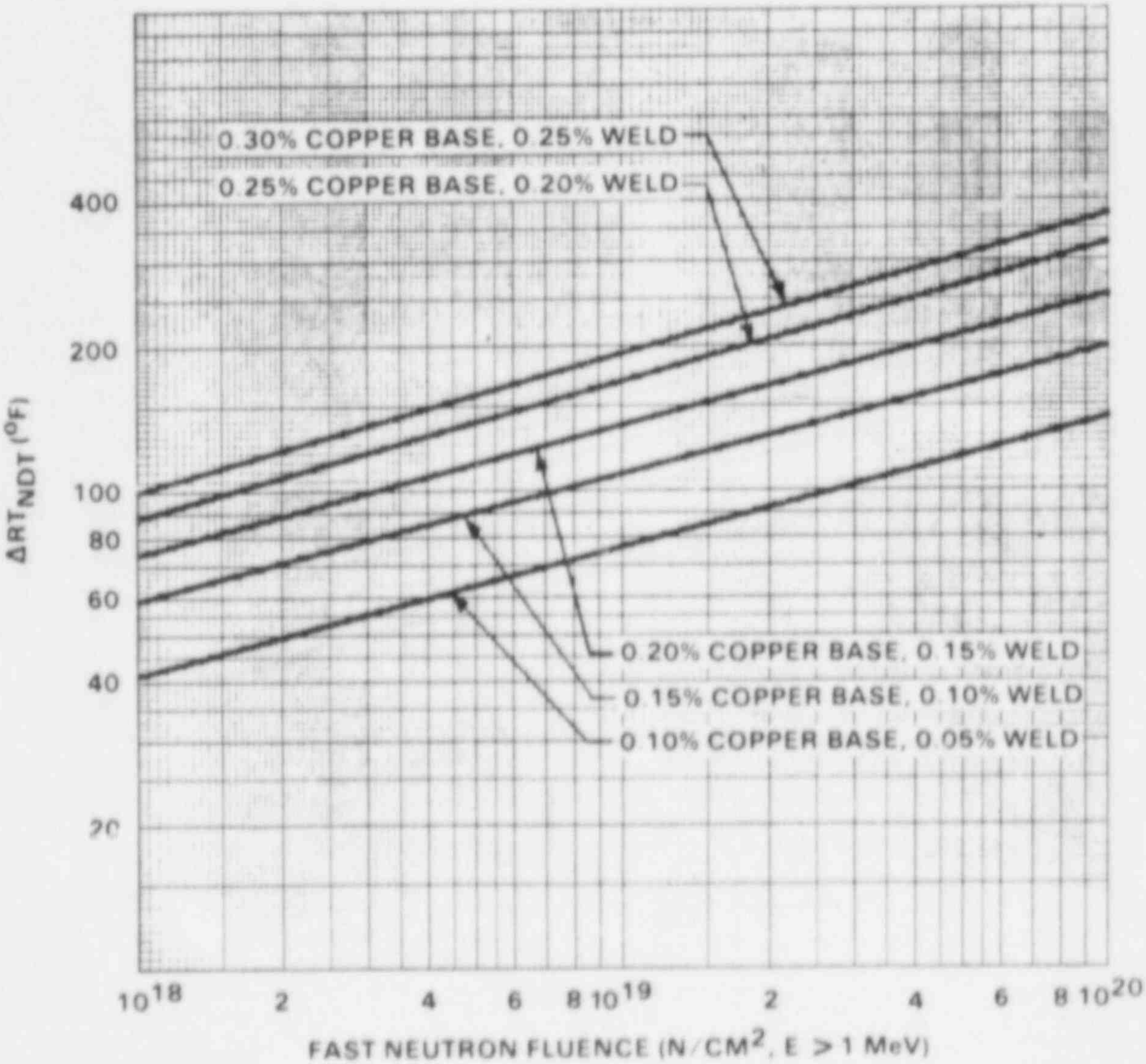


FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSELS EXPOSED TO 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T - RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

PRESSURE/TEMPERATURE LIMITS (Continued)

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T

PRESSURE/TEMPERATURE LIMITS (Continued)

defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

PRESSURE/TEMPERATURE LIMITS (Continued)

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or a Reactor Coolant System vent opening of at least 4.5 square inches ensures that the Reactor Coolant System will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the cold legs are less than or equal to 285°F. Either PORV has adequate relieving capability to protect the Reactor Coolant System from overpressurization when the transient is limited to either: (1) the start of an idle reactor coolant pump with the secondary water temperature of the steam generator less than or equal to 50°F above the cold leg temperatures, or (2) the start of a Safety Injection pump and its injection into a water solid Reactor Coolant System.

The Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance of the LTOPS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one Safety Injection pump and all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of a RCP if secondary temperature is more than 50°F above primary temperature.

The Maximum Allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head, and the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs from the cold leg injection accumulators and directly into the reactor vessel from the upper head injection accumulators in the event the Reactor Coolant System pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the coolant temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump and one Safety Injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE centrifugal charging pump to be inoperable below 285°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the refueling water storage tank and the Reactor Coolant System water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the refueling water storage tank also ensure a pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t; as applicable, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 1.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 15 psig during LOCA conditions.

INTERNAL PRESSURE (Continued)

The maximum peak pressure expected to be obtained from a LOCA event is 14.68 psig. The limit of 0.3 psig for initial positive containment pressure is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that: (1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions, and (2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment. Measurements shall be made at all operating ventilation unit locations, whether by fixed or portable instruments, prior to determining the average air temperature.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 100°F for the lower compartment and 75°F (60°F when in MODE 2, 3 or 4) for the upper compartment will limit the peak pressure to 14.7 psig which is less than the containment design pressure of 15 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the safety analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 15 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 REACTOR BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment reactor building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide: (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

3/4.6.1.8 ANNULUS VENTILATION SYSTEM

The OPERABILITY of the Annulus Ventilation System ensures that during LOCA conditions, containment vessel leakage into the annulus will be filtered through the HEPA filters and carbon adsorber trains prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. This requirement is necessary to meet the assumptions used in the safety analyses and limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during LOCA conditions. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.6.1.9 CONTAINMENT PURGE SYSTEMS

The containment purge supply and exhaust isolation valves for the lower compartment and the upper compartment (24-inch), and instrument room (12-inch), and the Hydrogen Purge System (4-inch) are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves sealed closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 4-inch Containment Air Release and Addition System valves since, unlike the lower compartment and the upper compartment, instrument room, and the Hydrogen Purge System valves, these 4-inch valves are capable of closing during a LOCA. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation with the line open will be limited to 2000 hours during a calendar year for the 4-inch valves. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to support the additional time requests.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere, and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with those assigned other inoperable ESF equipment.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations Following a LOCA," March 1971.

The OPERABILITY of at least 35 to 36 igniters per train (70 of 72 for both trains) ensures that the Distributed Ignition System will maintain an effective coverage throughout the containment provided the two inoperable ignitors are not on corresponding redundant circuits which provide coverage for the same region. This system of igniters will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 14.7 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will: (1) be distributed evenly through the containment bays, (2) contain sufficient boron to preclude dilution of the containment sump following the LOCA, and (3) contain sufficient heat removal capability to condense the Reactor Coolant System volume released during a LOCA. These conditions are consistent with the assumptions used in the safety analyses.

The minimum weight figure of 1218 pounds of ice per basket contains a 10% conservative allowance for ice loss through sublimation which is a factor of 10 higher than assumed for the ice condenser design. The minimum total weight of 2,368,652 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in the weighing instruments. In the event that observed sublimation rates are equal to or lower than design predictions after 3 years of operation, the minimum ice baskets weight may be adjusted downward. In addition, the number of ice baskets required to be weighed each 9 months may be reduced after 3 years of operation if such a reduction is supported by observed sublimation data.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the Ice Bed Temperature Monitoring System ensures that the capability is available for monitoring the ice temperature. In the event the system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.3 ICE CONDENSER DOORS

The OPERABILITY of the ice condenser doors and the requirement that they be maintained closed ensures that the Reactor Coolant System fluid released during a LOCA will be diverted through the ice condenser bays for heat removal and that excessive sublimation of the ice bed will not occur because of warm air intrusion.

If an Ice Condenser Door is not capable of opening automatically then system function is seriously degraded and immediate action must be taken to restore the opening capability of the door. Not capable of opening automatically is defined as those conditions in which a door is physically blocked from opening by installation of a blocking device or by obstruction from temporarily or permanently installed equipment. Impairment by ice, frost or debris is considered to render the doors inoperable but capable of opening automatically since these types of conditions will result in a slightly greater torque necessary to open the doors or a slight delay in door opening.

3/4.6.5.4 INLET DOOR POSITION MONITORING SYSTEM

The OPERABILITY of the Inlet Door Position Monitoring System ensures that the capability is available for monitoring the individual inlet door position. In the event the system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specfied time limits.

3/4.6.5.5 DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

The requirements for the divider barrier personnel access doors and equipment hatches being closed and OPERABLE ensure that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of the steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.6.5.6 CONTAINMENT AIR RETURN AND HYDROGEN SKIMMER SYSTEMS

The OPERABILITY of the Containment Air Return and Hydrogen Skimmer Systems ensures that following a LOCA: (1) the containment atmosphere is circulated for cooling by the spray system, and (2) the accumulation of hydrogen in localized portions of the containment structure is minimized. Since these systems are required to function during post-accident situations, the air density that the fans experience during surveillance testing will be different than the air density following a LOCA. An air density adjustment will be made to the test data before it is compared to the Technical Specification Surveillance Requirements.

3/4.6.5.7 and 3/4.6.5.8 FLOOR AND REFUELING CANAL DRAINS

The OPERABILITY of the ice condenser floor and refueling canal drains ensures that following a LOCA, the water from the melted ice and Containment Spray System has access for drainage back to the containment lower compartment and subsequently to the sump. This condition ensures the availability of the water for long-term cooling of the reactor during the post-accident phase.

3/4.6.5.9 DIVIDER BARRIER SEAL

The requirement for the divider barrier seal to be OPERABLE ensures that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1304 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from valve wide-open condition coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 16.85×10^6 lbs/h which is 105% of the total secondary steam flow of 16.05×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a feedwater line break accident with a worst case single active failure.

The Auxiliary Feedwater System is capable of delivering a total feedwater flow of at least 492 gpm at a pressure of 1210 psig to the entrance of at least two of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.4 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 CONDENSATE STORAGE SYSTEM

The OPERABILITY of the Condensate Storage System with the minimum water volume ensures that sufficient water is available to maintain the Reactor Coolant system at HOT STANDBY conditions for 2 hours followed by approximately 5 hours cooldown with steam discharge to the atmosphere concurrent with total loss-of-offsight power. The contained water volume limit includes an allowance for water not usuable because of tank discharge line location or other physical characteristics.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

The OPERABILITY of the Nuclear Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analysis.

3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND

The limitations on the standby nuclear service water pond level and temperature ensure that sufficient cooling capacity is available to either: (1) provide normal cooldown of the facility, or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 CONTROL ROOM AREA VENTILATION SYSTEM

The OPERABILITY of the Control Room Area Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.7.7 AUXILIARY BUILDING FILTERED EXHAUST SYSTEM

The OPERABILITY of the Auxiliary Building Filtered Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the auxiliary building following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Catawba Safety Review Group. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletions of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

SNUBBERS (Continued)

The visual inspection frequency is based upon maintaining a constant level of snubber protection during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. In order to establish the inspection frequency for each type of snubber, it was assumed that the frequency of snubber failures and initiating events are constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule. The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers.

To provide assurance of snubber functional reliability, one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life testing was performed to qualify the snubber for the applicable design conditions. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

SNUBBERS (Continued)

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System consists of the water system, spray, and/or sprinklers, CO_2, and fire hose stations. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility Fire Protection Program.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

FIRE SUPPRESSION SYSTEMS (Continued)

In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.11 FIRE BARRIER PENETRATIONS

The functional integrity of the fire barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. These design features minimize the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishing of the fire. The fire barrier penetrations are a passive element in the facility fire protection program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors, fire dampers, and other fire barriers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established, until the barrier is restored to functional status.

3/4.7.12 GROUNDWATER LEVEL

This specification is provided to ensure that groundwater levels will be monitored and prevented from rising to unacceptable levels. High groundwater levels could result in unacceptable structural stresses in the Containment and/or Auxiliary Building due to uplift and hydrostatic forces during design basis events. Although these buildings have been statically analyzed to withstand soil pressure along with the uplift and hydrostatic forces resulting from groundwater rebound to yard elevation (593'6"), this analysis did not include any other loadings and was not a design condition for these buildings.

3/4.7.13 STANDBY SHUTDOWN SYSTEM

The Standby Shutdown System (SSS) is designed to mitigate the consequences of certain postulated fire incidents by providing capability to maintain HOT STANDBY conditions and by controlling and monitoring vital systems from locations external to the main control room. This capability is consistent with the requirements of 10 CFR Part 50, Appendix R.

The Surveillance Requirements ensure that the SSS systems and components are capable of performing their intended functions. The required level in the SSS diesel generator fuel storage tank ensures sufficient fuel for 72 hours uninterrupted operation. It is assumed that, within 72 hours, either offsite power can be restored or additional fuel can be added to the storage tank.

Although the Standby Makeup Pump is not nuclear safety-related and was not designed according to ASME code requirements, it is tested quarterly to ensure its OPERABILITY. The Surveillance Requirement concerning the Standby Makeup Pump water supply ensures that an adequate water volume is available to supply the pump continuously for 72 hours.

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, Regulatory Guide 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979, and the NRC Staff Evaluation Report concerning the Reliability of Diesel Generators at Catawba, August 14, 1984. If any other metallic structures (building, new or modified piping systems, conduits) are placed in the ground near the Fuel Oil Storage System or if the original system is modified, the adequacy and frequency of inspections for the Cathodic Protection System shall be reevaluated and adjusted in accordance with the manufacturer's recommendations.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirements for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-3 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-3 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers and fuses provide assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker and/or fuse. Each manufacturer's molded case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuse for surveillance purposes.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The locking closed of the required valves during refueling operation precludes the possibility of uncontrolled boron dilution of the filled portion of the Reactor Coolant System. This action prevents flow to the Reactor Coolant System of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY of the Reactor Building Containment Purge System ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and carbon adsorbers prior to release to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the Reactor Building Containment Purge System and the resulting iodine removal capacity are consistent with the assumption of the safety analysis. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: (1) manipulator cranes will be used for movement of control rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two residual heat removal loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating residual heat removal loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating residual heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 and 3/4.9.10 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.11 FUEL HANDLING VENTILATION EXHAUST SYSTEM

The limitations on the Fuel Handling Ventilation Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and carbon adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure control rod worth, and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 and the Reactor Coolant System T_{avg} may fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Annal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of

DOSE (Continued)

Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks included in this specification are all those outdoor radwaste tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.1.5 CHEMICAL TREATMENT PONDS

The inventory limits of the chemical treatment ponds (CTP) are based on limiting the consequences of an uncontrolled release of the pond inventory. The expression in Specification 3.11.1.5 assumes the pond inventory is uniformly mixed, that the pond is located in an uncontrolled area as defined in 10 CFR Part 20, and that the concentration limit in Note 1 to Appendix B of 10 CFR Part 20 applies.

The batch limits of the resin/water slurry transferred to the CTP assure that radioactive material transferred to the CTP are "as low is reasonably achievable" in accordance with 10 CFR 50.36a. The expression in Specification 4.11.1.5 assures no batch will be transferred to the CTP unless the sum of the ratios of the activity of the radionuclides to their respective concentration limitation is less than the ratio of the 10 CFR Part 50, Appendix I, Section II.A, total body dose level to the 10 CFR 20.105(a), whole body dose limitation, or that:

$$\sum_j \frac{c_j}{C_j} < \frac{3\ \text{mrem/yr}}{500\ \text{mrem/yr}} = 0.006$$

Where:

c_j = radioactive resin/water slurry concentration for radionuclide "j" entering the UNRESTRICTED AREA CTP, in microCuries/milliliter; and

C_j = 10 CFR Part 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", in microCuries/milliliter.

CHEMICAL TREATMENT PONDS (Continued)

The filter/demineralizers using powdered resin and the blowdown demineralizer are backwashed or sluiced to a holding tank. The tank will be agitated to obtain a representative sample of the resin inventory in the tank. A known weight of the wet, drained resin (moisture content approximately 55 to 60%, bulk density of about 58 pounds per cubic foot) will then be counted. The concentration of the resin slurry to be pumped to the chemical treatment ponds will then be determined by the formula:

$$c_j = \frac{Q_j W_R}{V_T}$$

Where:

Q_j = concentration of radioactive materials in wet, drained resin for radionuclide "j", excluding tritium, dissolved or entrained noble gases, and radionuclides with less than an 8-day half-life. The analysis shall include at least Ce-144, Cs-134, Cs-137, Co-58 and Co-60, in microCuries/gram. Estimates of the Sr-89 and Sr-90 batch concentration shall be included based on the most recent monthly composite analysis (within 3 months);

W_R = total weight of resin in the storage tank in grams (determined from chemistry logs procedures); and

V_T = total volume of resin water mixture in storage tank to be transferred to the chemical treatment ponds in milliliters.

The batch limits provide assurance that activity input to the CTP will be minimized, and a means of identifying radioactive material in the inventory limitation of Specification 3.11.1.5.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion

DOSE RATE (Continued)

factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

The required detection capabilities for radioactive material in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

DOSE - NOBLE GASES (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactives waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of the calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 10 CFR 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/SOLIDIFICATION agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

TOTAL DOSE (Continued)

For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units and from outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ODCM are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e, similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

BASES

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4.

The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel surrounded by a concrete containment having the following design features:

a. Containment Vessel

1) Nominal inside diameter = 115 feet.
2) Nominal inside height = 171 feet.
3) Nominal thickness of vessel walls = 0.75 inch.
4) Nominal thickness of vessel dome = 0.6875 inch.
5) Nominal thickness of vessel bottom = 0.25 inch.
6) Net free volume = 1.2×10^6 cubic feet.

b. Reactor Building

1) Nominal Annular space = 6 feet.
2) Annulus nominal volume = 484,090 cubic feet.
3) Nominal outside height (top of foundation base to top of dome) = 177 feet.
4) Nominal inside diameter = 127 feet.
5) Minimum cylinder wall thickness = 3 feet.
6) Minimum dome thickness = 2.25 feet.
7) Dome inside radius = 87 feet.

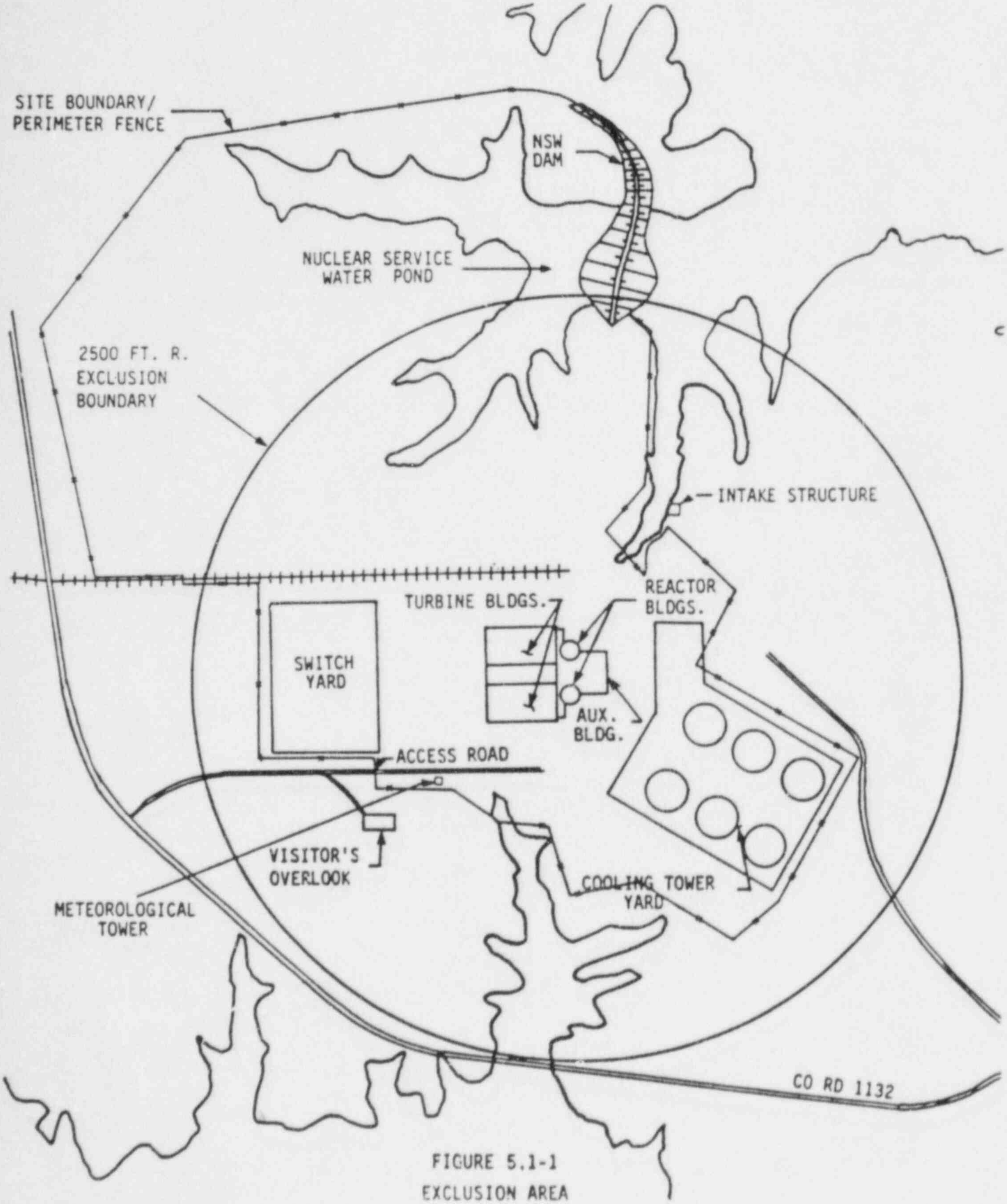


FIGURE 5.1-1
EXCLUSION AREA

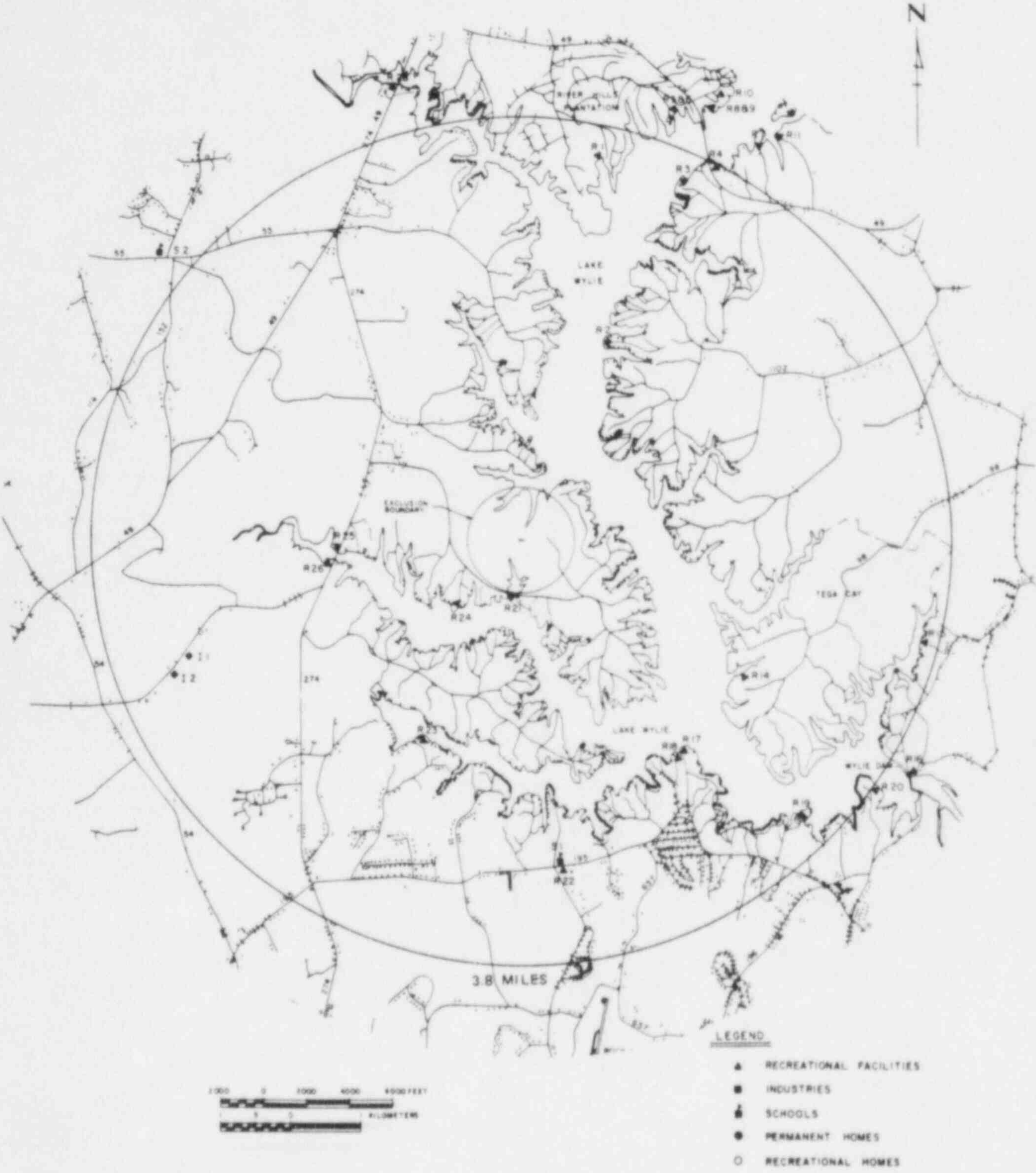


FIGURE 5.1-2
LOW POPULATION ZONE

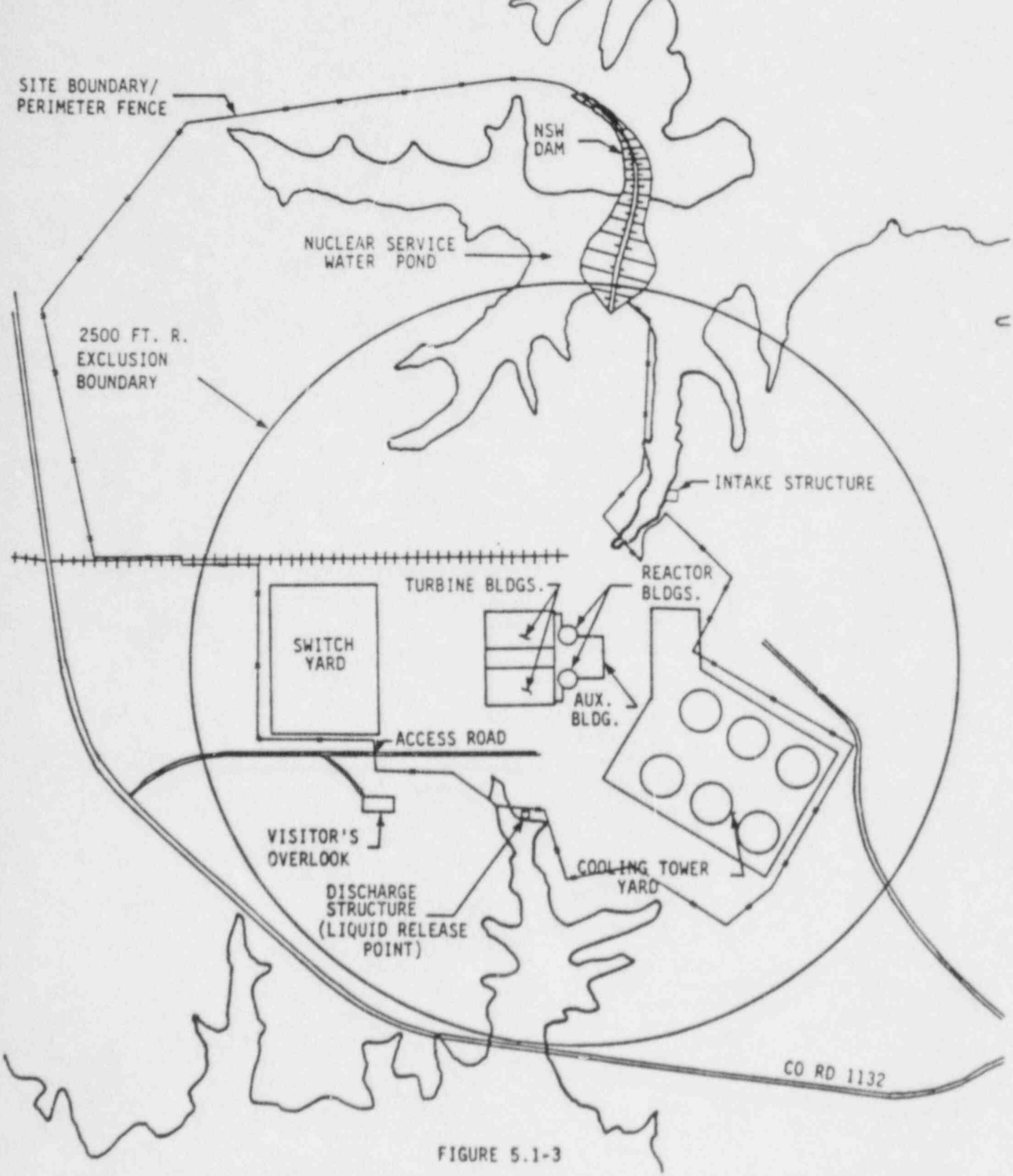


FIGURE 5.1-3

UNRESTRICTED AREA AND SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

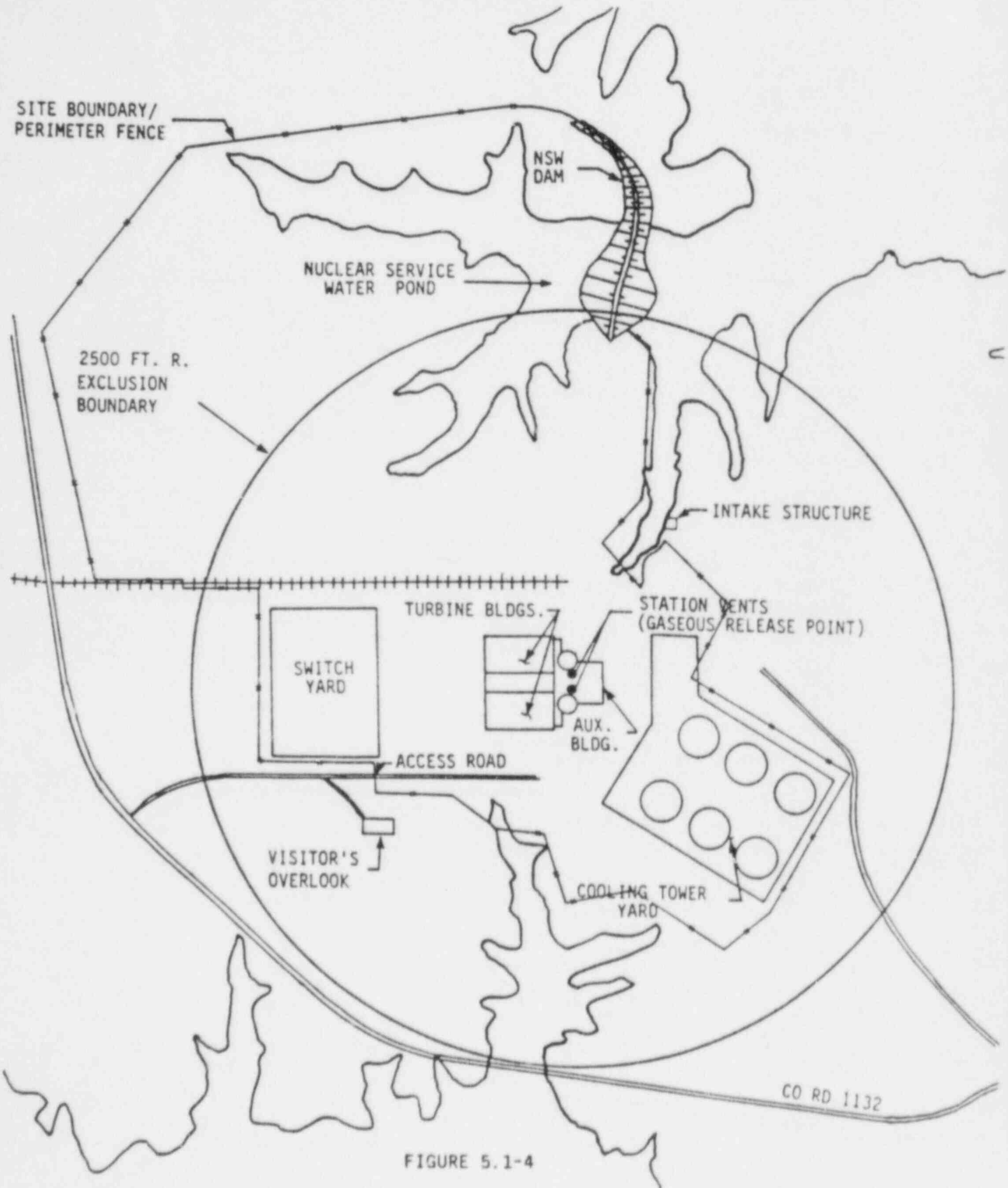


FIGURE 5.1-4

UNRESTRICTED AREA AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS EFFLUENTS

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment vessel is designed and shall be maintained for a maximum internal pressure of 15 psig and a temperature of 328°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1619 grams uranium. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material of which 102 inches shall be 100% boron carbide and remaining 40-inch tip shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure a 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,040 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown in Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent 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, which includes a conservative allowance for uncertainties as described in Section 9.1.2.3.1 of the FSAR, and

b. A nominal 21-inch for new fuel and a nominal 13.5-inch for spent fuel center-to-center distance between fuel assemblies placed in the storage racks.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 596 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1418 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER
	10 auxiliary spray actuation cycles.	Spray water temperature differential $>$ 320°F.
	50 leak tests.	Pressurized to $\geq$ 2485 psig.
	5 hydrostatic pressure tests.	Pressurized to $\geq$ 3106 psig.
Secondary Coolant System	1 steam line break.	Break in a $>$ 6-inch steam line.
	5 hydrostatic pressure tests.	Pressurized to $\geq$ 1350 psig.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President of Nuclear Production, shall be reissued to all Nuclear Production Department station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator for each unit shall be in the control room when fuel is in the reactor. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. A Health Physics Technician* shall be on site when fuel is in either reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include the three members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Health Physics Technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Station Manager or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Station Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

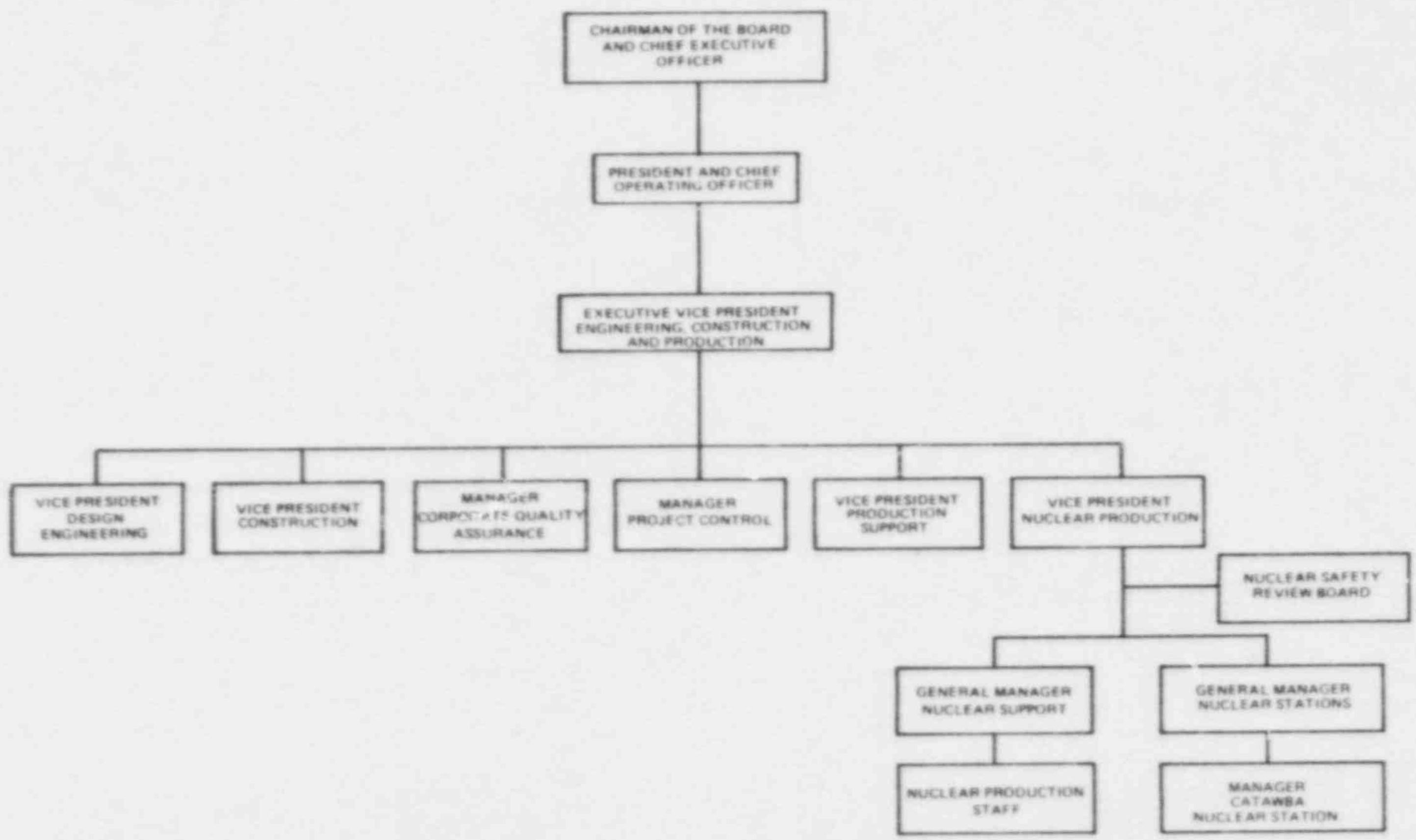
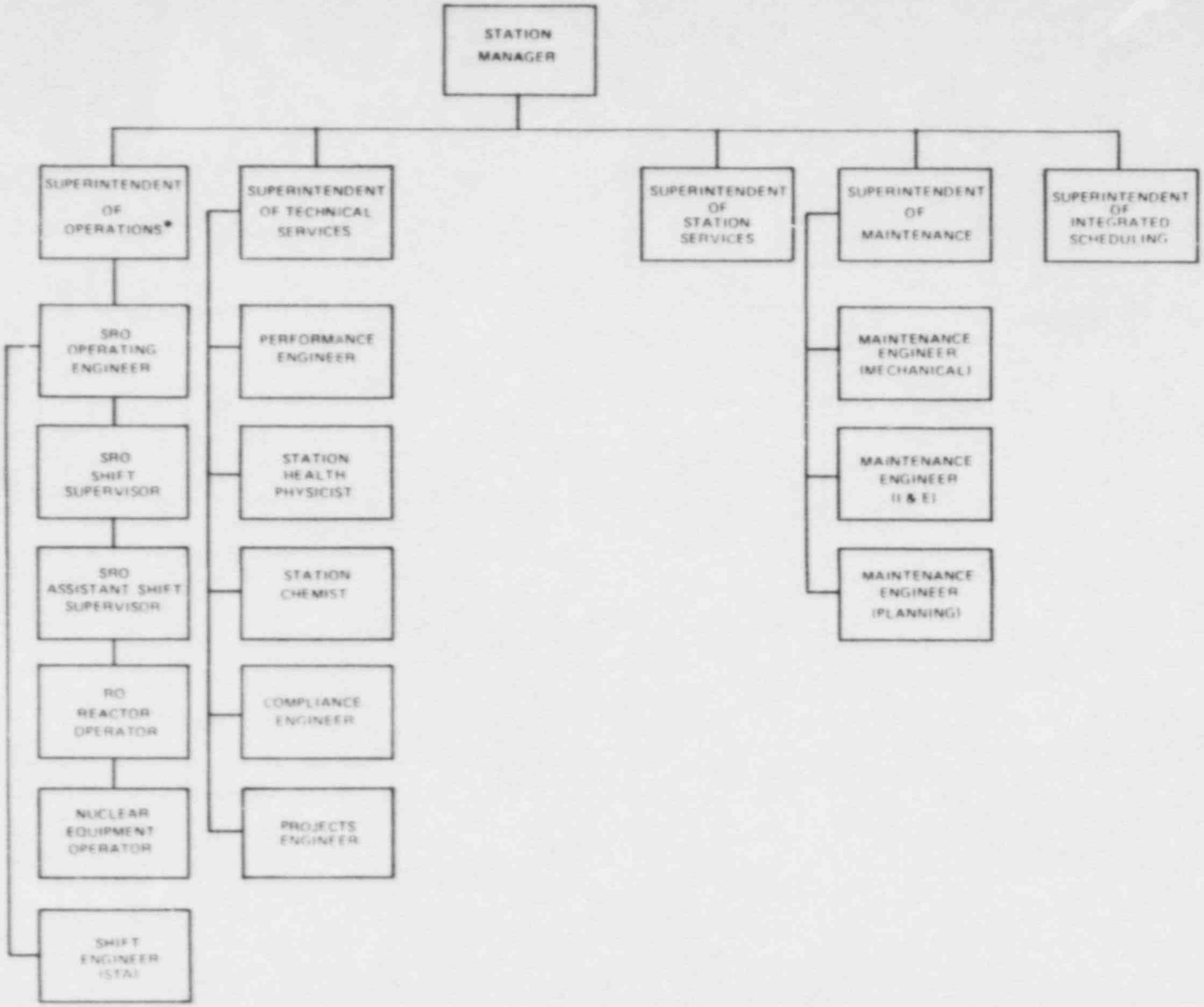


FIGURE 6.2-1

OFFSITE ORGANIZATION



LEGEND
SRO - SENIOR REACTOR OPERATOR
RO - REACTOR OPERATOR
*Superintendent of Operations must hold or have held SRO License

FIGURE 6.2-2

UNIT ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	Both Units in MODE 1, 2, 3, or 4	Both Units in MODE 5 or 6 or Defueled	One Unit in Mode 1, 2, 3 or 4 and One Unit in Mode 5 or 6 or Defueled
SS	1	1	1
SRO	1	None##	1
RO	3#	2#	3#
NEO	3#	3#	3#
STA	1	None	1

SS - Shift Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
NEO - Nuclear Equipment Operator
STA - Shift Technical Advisor

The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor*) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

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*On occasion when there is a need for both the Shift Supervisor and the SRO to be absent from the control room, the STA shall be allowed to assume the control room command function and serve as the SRO in the control room provided that: (1) the Shift Supervisor is available to return to the control room within 10 minutes, (2) the assumption of SRO duties by the STA be limited to periods not in excess of 15 minutes duration and a total time not to exceed 1 hour during any 8-hour shift, and (3) the STA has a Senior Operator license on the unit.

#At least one of the required individuals must be assigned to the designated position for each unit.

##At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

6.2.3 CATAWBA SAFETY REVIEW GROUP

FUNCTION

6.2.3.1 The Catawba Safety Review Group (CSRG) shall function to examine plant operating characteristics, NRC issuances, industry advisories, REPORTABLE EVENTS, and other sources which may indicate areas for improving plant safety. The CSRG shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to the Director, Nuclear Safety Review Board.

COMPOSITION

6.2.3.2 The CSRG shall be composed of a chairman and at least four dedicated, full-time qualified individuals located onsite.

RESPONSIBILITIES

6.2.3.3 The CSRG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the CSRG shall be prepared, maintained, and forwarded each calendar month to the Director, Nuclear Safety Review Board.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Supervisor.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiation Protection Manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.**

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Station Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the CSRG.

* Not responsible for sign-off function.

** Except that the experience and other considerations described in Duke Power Company's letter dated 28 August 1985 are acceptable for Unit 1 for the six applicants for SRO licenses identified therein.

and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the CSRG.

6.5 REVIEW AND AUDIT

6.5.1 TECHNICAL REVIEW AND CONTROL ACTIVITIES

6.5.1.1 Each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, shall be prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.1.2 Proposed changes to the Appendix A Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specification change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager.

6.5.1.3 Proposed modifications to unit nuclear safety-related structures, systems, and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems, and components shall be approved prior to implementation by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent, or the Maintenance Superintendent, as previously designated by the Station Manager.

6.5.1.4 Individuals responsible for reviews performed in accordance with Specifications 6.5.1.1, 6.5.1.2, and 6.5.1.3 shall be members of the station supervisory staff, previously designated by the Station Manager to perform such reviews. Review of environmental radiological analysis procedures shall be performed by the Corporate System Health Physicist or his designee. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated station review personnel.

6.5.1.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager; or by the Operating Superintendent, the Technical Services Superintendent, or the Maintenance Superintendent, as previously designated by the Station Manager.

TECHNICAL REVIEW AND CONTROL ACTIVITIES (Continued)

6.5.1.6 All REPORTABLE EVENTS and all violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production, and to the Director of the Nuclear Safety Review Board.

6.5.1.7 The Station Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President, Nuclear Production.

6.5.1.8 The station security program, and implementing procedures shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production, and to the Director of the Nuclear Safety Review Board.

6.5.1.9 The station emergency plan, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager and transmitted to the Vice President, Nuclear Production, and to the Director of the Nuclear Safety Review Board.

6.5.1.10 The Station Manager shall assure the performance of a review by a qualified individual/organization of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective ACTION to prevent recurrence to the Vice President, Nuclear Production and to the Nuclear Safety Review Board.

6.5.1.11 The Station Manager shall assure the performance of a review by a qualifed individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and Radwaste Treatment Systems.

6.5.1.12 Reports documenting each of the activities performed under Specifications 6.5.1.1 through 6.5.1.11 shall be maintained. Copies shall be provided to the Vice President, Nuclear Production, and the Nuclear Safety Review Board.

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB)

FUNCTION

6.5.2.1 The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

FUNCTION (Continued)

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Administrative control and quality assurance practices.

The NSRB shall report to and advise the Vice President, Nuclear Production, on those areas of responsibility specified in Specifications 6.5.2.8 and 6.5.2.9.

ORGANIZATION

6.5.2.2 The Director, members, and alternate members of the NSRB shall be appointed in writing by the Vice President, Nuclear Production, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of 5 years technical experience, of which a minimum of 3 years shall be in one or more areas given in Specification 6.5.2.1. No more than two alternates shall participate as voting members in NSRB activities at any one time.

6.5.2.3 The NSRB shall be composed of at least five members, including the Director. Members of the NSRB may be from the Nuclear Production Department, from other departments within the Company, or from external to the Company. A maximum of one member of the NSRB may be from the Catawba Nuclear Station staff.

6.5.2.4 Consultants shall be utilized as determined by the NSRB Director to provide expert advice to the NSRB.

6.5.2.5 Staff assistance may be provided to the NSRB in order to promote the proper, timely, and expeditious performance of its functions.

6.5.2.6 The NSRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

6.5.2.7 The quorum of the NSRB necessary for the performance of the NSRB review and audit functions of these Technical Specifications shall consist of the Director, or his designated alternate, and at least four other NSRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of Catawba Nuclear Station.

REVIEW

6.5.2.8 The NSRB shall be responsible for the review of:

a. The safety evaluation for: (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

d. Proposed changes in Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety;

i. Quality Assurance Department audits relating to station operations and actions taken in response to these audits; and

j. Reports of activities performed under the provisions of Specifications 6.5.1.1 through 6.5.1.11.

AUDITS

6.5.2.9 Audits of unit activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training, and qualifications of the entire unit staff at least once per 12 months;

AUDITS (Continued)

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The Emergency Plan and implementing procedures at least once per 12 months;

f. The Security Plan and implementing procedures at least once per 12 months;

g. The Facility Fire Protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

h. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

i. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

j. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

k. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

l. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

m. Any other area of unit operation considered appropriate by the NSRB or the Vice President, Nuclear Production.

RECORDS

6.5.2.10 Records of NSRB activites shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRB meeting shall be prepared, approved, and forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Power Operations, within 14 days following each meeting;

RECORDS (Continued)

b. Reports of reviews encompassed by Specification 6.5.2.8 above, shall be prepared, approved, and forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Power Operations, within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.9 above, shall be forwarded to the Vice President, Nuclear Production, and to the Executive Vice President, Power Operations, and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50 and

b. Each REPORTABLE EVENT shall be reviewed by the Station Manager; or by (1) Operating Superintendent; (2) Technical Services Superintendent; or (3) Maintenance Superintendent, as previously designated by the Station Manager, and the results of this review shall be submitted to the NSRB and the Vice President-Nuclear Production.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear Production and the NSRB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Operating Superintendent and Station Manager. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence;

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Vice President-Nuclear Production within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement No. 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation; and

g. Quality Assurance Program implementation for effluent and environmental monitoring.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed and approved by the Station Manager; or by: (1) Operating Superintendent, (2) Technical Services Superintendent, or (3) Maintenace Superintendent, as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed, and approved by the Station Manager; or by: (1) Operating Superintendent, (2) Technical Services Superintendent, or (3) Maintenance Superintendent, as previously designated by the Station Manager; within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the containment spray, Safety Injection, chemical

PROCEDURES AND PROGRAMS (Continued)

and volume control, and nuclear 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. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Pr cedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective actions for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1) Training of personnel, and

2) Procedures for monitoring.

PROCEDURES AND PROGRAMS (Continued)

e. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT***

6.9.1.6 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by Specification 3.12.2.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.

***A single submittal may be made for a multiple unit station.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, and the corrective actions being taken if the specified program is not being performed as required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; reasons for not conducting the Radiological Environmental Monitoring Program as required by Specification 3.12.1 and discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

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*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

**A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Total Container volume,

b. Total Curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Sources of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, large quantity), and

f. Solidification agent or absorbent (e.g., cement urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous ycar. This annual summary may be either in the form on an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation". Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

*In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PCP and to the ODCM, pursuant to Specifications 6.13 and 6.14, respectively, as well as any major changes to Liquid, Gaseous or Solid Radwaste Treatment Systems, pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.9 The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the Regional Office of the NRC with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, for all core planes containing bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs Axial Core Height with the limit envelope at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new submittal or an amended submittal to the Radial Peaking Factor Limit Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the unit staff;

RECORD RETENTION (Continued)

h. Records of inservice inspections performed pursuant to these Technical Specifications;

i. Records of quality assurance activities required by the Operational Quality Assurance Manual;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the NSRB and reports required by Specification 6.5.1.10;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality; and

n. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

6.11 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 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 requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated

HIGH RADIATION AREA (Continued)

dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Station Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

    2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

    3) Documentation of the fact that the change has been reviewed and found acceptable by the Station Manager.

6.13 PROCESS CONTROL PROGRAM (PCP) (Continued)

b. Shall become effective upon review and acceptance by a qualified individual/organization.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

 1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated, and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

 2) A determination that the change will not reduce the accuracy or reliability of dose calculations of Setpoint determinations; and

 3) Documentation of the fact that the change has been reviewed and found acceptable by the Station Manager.

b. Shall become effective upon review and acceptance by a qualified individual/organization.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.15 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Station Manager. The discussion of each change shall contain:

 1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

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*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS (Continued)

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the Station Manager.

b. Shall become effective upon review and acceptance by a qualified individual/organization.

7.0 SPECIAL TEST PROGRAM

For conducting the Special Low Power Test Program as described in the Catawba FSAR Table 14.2.12-2, pages 35 and 36, the following Technical Specification requirements may be excepted:

Specification 2.2.1	Reactor Trips (Table 2.2-1)
	7. Overtemperature ΔT 8. Overpower ΔT
Specification 3.3.1	Reactor Trip System Instrumentation (Tables 3.3-1 and 3.3-2)
	7. Overtemperature ΔT 8. Overpower ΔT
Specification 3.5.1.2	Upper Head Injection Accumulator System

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16. ABSTRACT (200 words or less)

The Catawba Nuclear Station, Units 1 and 2 Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

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