NORTHEAST NUCLEAR ENERGY COMPANY*

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

NOTICE OF ISSUANCE OF FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission or NRC), has issued Facility Operating License No. NPF-49 to Northeast Nuclear Energy Company and 15 owners listed below* (the licensees) which authorizes operation of the Millstone Nuclear Power Station, Unit No. 3 (the facility), at reactor core power levels not in excess of 3411 megawatts thermal in accordance with the provisions of the License, the Technical Specifications and the Environmental Protection Plan. The issuance of this license was approved by the Nuclear Regulatory Commission at a meeting on January 29, 1986, and supersedes the License for Fuel Loading and Low Power Testing License NPF-44 issued on November 25, 1985.

The Millstone Nuclear Power Station, Unit No. 3 (Millstone Unit 3) is a pressurized water reactor located on the north shore of Long Island Sound in Waterford Township, New London County, Connecticut. The license is effective as of the date of issuance.

The application for the license complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the

---

* Northeast Nuclear Energy Company acts as agent and representative for the following Owners: Central Maine Power Company, Central Vermont Public Service Corporation, Chicopee Municipal Lighting Plant, City of Burlington, Vermont, Connecticut Municipal Electric Energy Cooperative, The Connecticut Light and Power Company, Fitchburg Gas and Electric Company, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, Public Service Company of New Hampshire, The United Illuminating Company, The Village of Lydonville Electric Department, Western Massachusetts Electric Company, and Vermont Electric Generation and Transmission Cooperative, Inc.

Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I which are set forth in the license. Prior public notice of the overall action involving the proposed issuance of an operating license was published in the Federal Register on March 4, 1983 (48 FR 9408).

The Commission has determined that the issuance of this license will not result in any environmental impacts other than those evaluated in the Final Environmental Statement since the activity authorized by the license is encompassed by the overall action evaluated in the Final Environmental Statement.

For further details with respect to this action, see (1) Facility Operating License No. NPF-49, with Technical Specifications (NUREG-1176) and the Environmental Protection Plan; (2) the report of the Advisory Committee on Reactor Safeguards, dated September 10, 1984; (3) the Commission's Safety Evaluation Report, dated July 1984 (NUREG-1031), and Supplements 1 through 5; (4) the Final Safety Analysis Report and Amendments thereto; (5) the Environmental Report and supplements thereto; and (6) the Final Environmental Statement dated December 1984 (NUREG-1064).

These items are available for inspection at the Commission's Public Document Room located at 1717 H Street, N.W., Washington, D. C. 20555 and in the Waterford Public Library, Rope Ferry Road, Route 156, Waterford, Connecticut 06385. A copy of Facility Operating License NPF-49 may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Project Director, PWR Licensing-A. Copies of the Safety Evaluation Report and Supplements 1 through 5 (NUREG-1031) and the

Final Environmental Statement (NUREG-1064) may be purchased at current rates from the Superintendent of Documents, U. S. Government Printing Office, Post Office Box 37082, Washington D. C. 20013-7982 or by calling (202) 275-2060 or (202) 275-2171.

Dated at Bethesda, Maryland this 31st day of January 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Vincent S. Noonan, Director
PWR Project Directorate #5
Division of PWR Licensing-A

Final Environmental Statement (NUREG-1064) may be purchased at current rates from the Superintendent of Documents, U. S. Government Printing Office, Post Office Box 37082, Washington D. C. 20013-7982 or by calling (202) 275-2060 or (202) 275-2171.

Dated at Bethesda, Maryland this 31st day of January, 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Vincent S. Noonan, Director
PWR Project Directorate #5
Division of PWR Licensing-A

PWR 5	PWR 5	OELD	PWR5
MRushbrook/mac	EDoolittle		VSNoonan
1/9/85	1/19/85	1/28/85	1/30/85
		J. Scinto	

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

Docket Nos. 50-245
50-336
50-423

AMENDMENT TO INDEMNIT\ AGREEMENT NO. B-39
AMENDMENT NO. 18

Effective January 31, 1986, Indemnity Agreement No. B-39, between Northeast Nuclear Energy Company, The Connecticut Light and Power Company, Western Massachusetts Electric Company, New England Power Company, United Illuminating Company, Public Service Company of New Hampshire, Central Vermont Public Service Corporation, Montaup Electric Company, City of Burlington, Vermont, Electric Light Department, Chicopee Municipal Lighting Plant, Massachusetts Municipal Wholesale Electric Company, Vermont Electric Generation and Transmission Cooperative, Inc., Central Maine Power Company, Village of Lyndonville Electric Department, Connecticut Municipal Electric Energy Cooperative, Fitchburg Gas and Electric Light Company and the Atomic Energy Commission, dated May 9, 1969, as amended, is hereby further amended as follows:

Item 3 of the Attachment to the indemnity agreement is deleted in its entirety and the following substituted therefor:

Item 3 - License number or numbers

SNM-1098	(From 12:01 a.m., May 9, 1969, to 12 midnight, October 6, 1970 inclusive)
DPR-21	(From 12:01 a.m., October 7, 1970)
SNM-1335	(From 12:01 a.m., January 22, 1973, to 12 midnight, July 31, 1975 inclusive)
DPR-65	(From 12:01 a.m., August 1, 1975)
SNM-1950	(From 12:01 a.m., April 16, 1985, to 12 midnight, November 24, 1985 inclusive)
NPF-44	(From 12:01 a.m., November 25, 1985, to 12 midnight, January 30, 1986 inclusive)

By______________________________	By______________________________
CENTRAL MAINE POWER COMPANY	VILLAGE OF LYNDONVILLE ELECTRIC DEPARTMENT
Accepted__________________, 1985	Accepted__________________, 1985
By______________________________	By______________________________
CONNECTICUT MUNICIPAL ELECTRIC ENERGY COOPERATIVE	FITCHBURG GAS AND ELECTRIC LIGHT COMPANY

NPF-49 (From 12:01 a.m., January 31, 1986)

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Darrel A Nash for

Jerome Saltzman, Assistant Director
State and Licensee Relations

Office of State Programs

Accepted ________________, 1985

By________________
NORTHEAST NUCLEAR ENERGY COMPANY

Accepted ________________, 1985

By________________
THE CONNECTICUT LIGHT AND POWER COMPANY

Accepted ________________, 1985

By________________
WESTERN MASSACHUS[illegible]S ELECT[illegible] COMPANY

Accepted ________________, 1985

By________________
NEW ENGLAND POWER COMPANY

Accepted ________________, 1985

By________________
UNITED ILLUMINATING COMPANY

Accepted ________________, 1985

By________________
PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE

Accepted ________________, 1985

By________________
CENTRAL VERMONT PUBLIC SERVICE CORPORATION

Accepted ________________, 1985

By________________
MONTAUP ELECTRIC COMPANY

Accepted ________________, 1985

By________________
CITY OF BURLINGTON, VERMONT, ELECTRIC LIGHT DEPARTMENT

Accepted ________________, 1985

By________________
CHICOPEE MUNICIPAL LIGHTING PLANT

Accepted ________________, 1985

By________________
MASSACHUSETTS MUNICIPAL WHOLESALE ELECTRIC COMPANY

Accepted ________________, 1985

By________________
VERMONT ELECTRIC GENERATION AND TRANSMISSION COOPERATIVE, INC.

Accepted ________________, 1985

NUREG-1176

Technical Specifications
Millstone Nuclear Power Station, Unit No. 3

Docket No. 50-423

Appendix "A" to
License No. NPF-49

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

January 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-1176

Technical Specifications
Millstone Nuclear Power Station, Unit No. 3

Docket No. 50-423

Appendix "A" to
License No. NPF-49

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

Office of Nuclear Reactor Regulation

January 1986

INDEX

DEFINITIONS

DEFINITIONS

SECTION	PAGE

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3 INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.6 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.7 PLANT SYSTEMS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

BASES

BASES

SECTION PAGE

BASES

SECTION PAGE

3/4.11 RADIOACTIVE EFFLUENTS

DESIGN FEATURES

ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

1.1 ACTION shall be that part of a 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 ALTERATIONS

1.9 CORE ALTERATIONS 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 ALTERATIONS 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 NRC Regulatory Guide 1.109, Revision 1, "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."

$\bar{E}$ - 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.

ENCLOSURE BUILDING INTEGRITY

1.12 ENCLOSURE 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,

b. The Supplementary Leak Collection and Release System is OPERABLE, and

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

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FIRE DETECTOR OPERATIONAL TEST

1.14 A FIRE DETECTOR OPERATIONAL TEST shall be the injection of a simulated signal at the primary sensor to verify detector OPERABILITY and alarm transmission to the local zone panel.

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

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

b. Leakage into the containment atmosphere from sources that are both specifically located and known 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

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

MEMBER(S) OF THE PUBLIC

1.18 MEMBER(S) OF THE PUBLIC 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.

The term "REAL MEMBER OF THE PUBLIC" means an individual who is exposed to existing dose pathways at one particular location.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

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.

RADIOACTIVE WASTE TREATMENT SYSTEMS

1.25 RADIOACTIVE WASTE TREATMENT SYSTEMS are those liquid, gaseous and solid waste systems which are required to maintain control over radioactive material in order to meet the Limiting Conditions for Operation (LCOs) set forth in these specifications.

RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

1.26 A RADIOLOGICAL EFFLUENT MONITORING MANUAL shall be a manual containing the site and environmental sampling and analysis programs for measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures to individuals from station operation. An OFFSITE DOSE CALCULATION MANUAL shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setponts. Requirements of the REMODCM are provided in Specification 6.16.

RATED THERMAL POWER

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

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

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

SLAVE RELAY TEST

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

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

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

VENTING

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

TABLE 1.1

FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 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 Figures 2.1-1 and 2.1-2 for four and three loop operation, respectively.

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 2750 psia.

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

ACTION:

MODES 1 and 2:

> Whenever the Reactor Coolant System pressure has exceeded 2750 psia be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

> Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

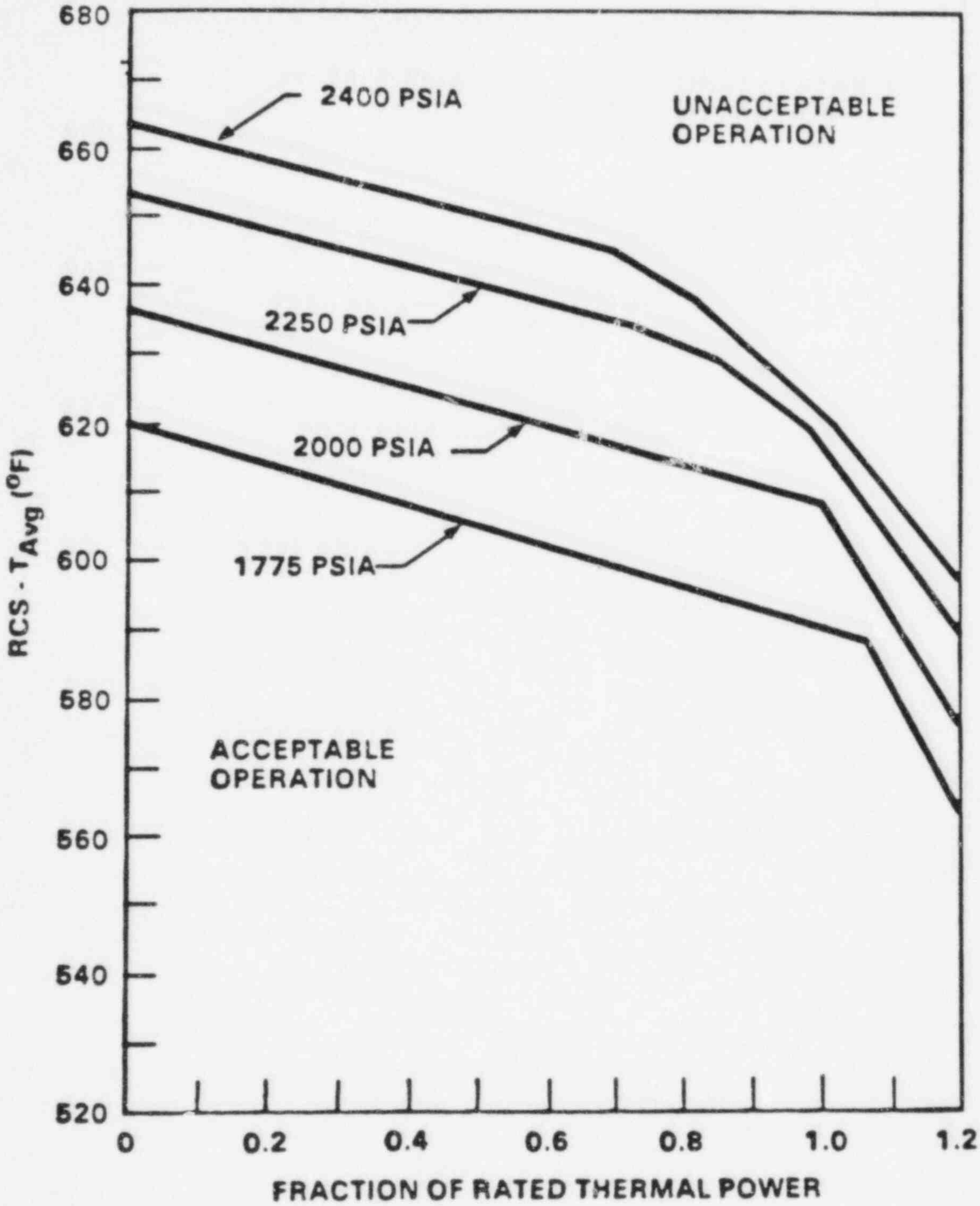


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

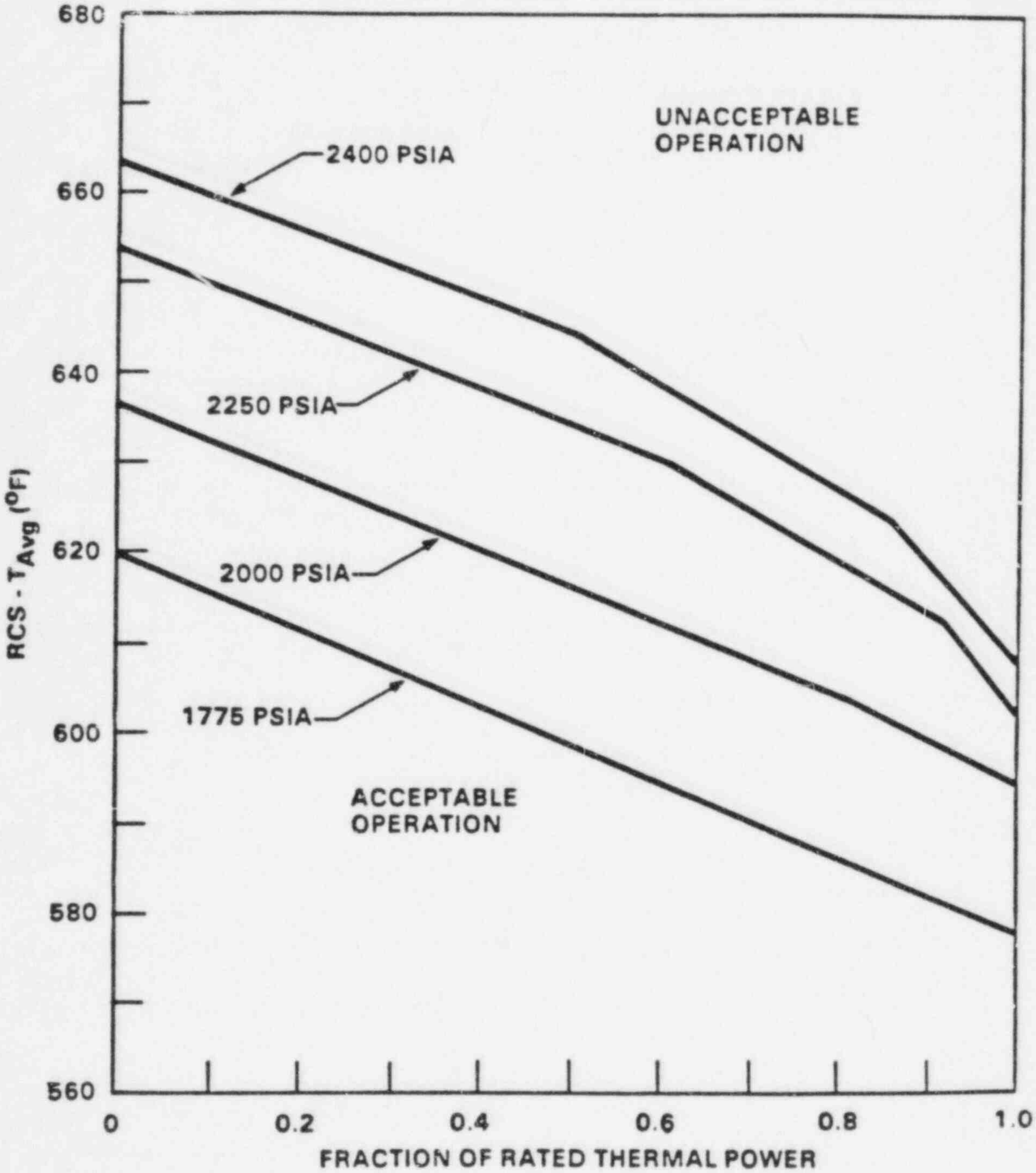


FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable 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 $$Z + R + S \leq TA$$

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					
1) Four Loops Operating	7.5	4.56	0	≤ 109% of RTP**	≤ 111.1% of RTP**
2) Three Loops Operating	7.5	4.56	0	≤ 80% of RTP**	≤ 82.1% 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	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT					
a. Four Loops Operating	8.3	5.9	2.3	See Note 1	See Note 2
b. Three Loops Operating	12.0	5.9	2.3	See Note 1	See Note 2
8. Overpower ΔT	4.8	1.43	0.11	See Note 3	See Note 4

*Loop design flow = 94,600 gpm (Four Loops Operating); 99,600 (Three Loops Operating)
**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low	5.0	1.77	3.3	≥ 1900 psia	≥ 1890 psia
10. Pressurizer Pressure-High	5.0	1.77	3.3	≤ 2385 psia	≤ 2395 psia
11. Pressurizer Water Level-High	8.0	5.13	2.7	≤ 89% of instrument span	≤ 90.7% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.74	0.8	≥ 90% of loop design flow*	≥ 89.3% of loop design flow*
13. Steam Generator Water Level Low-Low	20.5	18.98	1.75	≥ 23.5% of narrow range instrument span	≥ 22.6% of narrow range instrument span
14. General Warning Alarm	N.A.	N.A.	N.A.	N.A.	N.A.
15. Low Shaft Speed - Reactor Coolant Pumps	3.8	0.5	0	≥ 97.8% of rated speed	≥ 94.6% of rated speed
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥ 500 psig	≥ 450 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	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤ 10% of RTP**	≤ 12.1% of RTP**
2) P-13 input	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine Impulse Pressure Equivalent	≤ 12.1% RTP** Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8					
1) Four Loops Operating	N.A.	N.A.	N.A.	≤ 37.5% of RTP**	≤ 39.6% of RTP**
2) Three Loops Operating	N.A.	N.A.	N.A.	≤ 37.5% of RTP**	≤ 39.6% of RTP**
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤ 51% of RTP**	≤ 53.1% of RTP**
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	≥ 10% of RTP**	≥ 7.9% of RTP**
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.
21. Three Loop Operation Bypass Circuitry	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

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

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, τ_1 = 8 s, τ_2 = 3 s;

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

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.08 (Four Loops Operating); 1.01 (Three Loops Operating);

K_2 = 0.01313;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = 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}, τ_4 = 33 s, τ_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, τ_6 = 0 s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	587.1°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000663/psia;
P	=	Pressurizer pressure, psia;
P'	=	2250 psia (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:

(1) For $q_t - q_b$ between -30% and + 10%, $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -30%, the ΔT Trip Setpoint shall be automatically reduced by 3.6% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +10%, the ΔT Trip Setpoint shall be automatically reduced by 2.0% 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 2.1% ΔT span (Four Loop Operation); 4.1% ΔT span (Three Loop Operation).

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where:

ΔT	=	As defined in Note 1,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1,
τ_1, τ_2	=	As defined in Note 1,
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	=	1.09,
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 compensator for T_{avg} dynamic compensation,
τ_7	=	Time constants utilized in the rate-lag compensator 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	=	0.00129/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 587.1°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 3.4% ΔT span.

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 SAFETY LIMITS

BASES

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 W-3 R-Grid correlation. The W-3 R-Grid 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 minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 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:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$$

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 (RCS) 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 RCS 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% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3125 psia) of design pressure, to demonstrate integrity prior to initial operation.

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 and instrument drift.

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 safety 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 High Setpoint trip is reduced during three loop operation to a value consistent with the safety analysis.

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 multiple 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 1.30.

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. 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. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Trip System.

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 2.1-1. 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.

Operation with a reactor coolant loop out of service requires Reactor Trip System modification. Three loop operation is permissible after resetting the K1 input to the Overtemperature ΔT channels, reducing the Power Range Neutron Flux High setpoint to a value just above the three loop maximum permissible power level, and resetting the P-8 setpoint to its three loop value. These modifications have been chosen so that, in three loop operation, each component of the Reactor Trip System performs its normal four loop function, prevents operation outside the safety limit curves, and prevents the DNBR from going below 1.30 during normal operational and anticipated transients.

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, 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 Water Level High 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 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, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Flow Low trip provides core protection to prevent 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 38% 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 the 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.

Low Shaft Speed - Reactor Coolant Pumps

The Low Shaft Speed - Reactor Coolant Pumps trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant pump speed (with resulting decrease in flow) on two reactor coolant pumps in any two operating reactor coolant loops. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained above 1.30 following a four-pump loss of flow event.

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 50% 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) and 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 low shaft speed, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

Reactor Trip System Interlocks (Continued)

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.

Three Loop Operation Bypass Circuitry

The Three Loop Operation Bypass Circuitry reactor trip ensures that a sufficient number and the correct combination of trip circuits remain available to provide necessary protection and mitigation capability during three loop operation. Should more than two channels in one train or two dissimilar channels in two trains be bypassed, a reactor trip will occur. In this manner, it is ensured that sufficient protective features remain to mitigate the consequences of analyzed transients.

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.

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

LIMITING CONDITION FOR OPERATION (Continued)

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 has 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);

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;

LIMITING CONDITION FOR OPERATION (Continued)

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} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% $\Delta k/k$ for both four loop and three loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.6% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6300 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 1.6% $\Delta k/k$:

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 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 insertion 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% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., 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.

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.6% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6300 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% $\Delta k/k$:

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.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

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

b. Less negative than -4.0×10^{-4} $\Delta k/k/^{\circ}F$ for the all 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 $\Delta k/k/^{\circ}F$ 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 -3.1×10^{-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 -3.1×10^{-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 tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid storage system 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 Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature are greater than or equal to 67°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 (RCS), and

b. Two flow paths from the refueling water storage tank via charging pumps to the RCS.

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

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.6% $\Delta k/k$ 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 Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature are greater than or equal to 67°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 33 gpm to the RCS.

---

*Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°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, on recirculation flow, that a differential pressure across the pump of greater than or equal to 2411 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.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two* charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.6% $\Delta k/k$ 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, on recirculation flow, that a differential pressure across each pump of greater than or equal to 2411 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above allowed OPERABLE pump, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the Reactor Coolant System (RCS) cold legs is less than or equal to 350°F by verifying that the motor circuit breakers are secured in the open position.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°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 6700 gallons,

2) A boron concentration between 6300 and 7175 ppm, and

3) A minimum solution temperature of 67°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 250,000 gallons,

2) A minimum boron concentration of 2000 ppm, and

3) A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 35°F.

REACTIVITY CONTROL SYSTEMS

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 23,620 gallons,

2) A boron concentration between 6300 and 7175 ppm, and

3) A minimum solution temperature of 67°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 1,166,000 gallons,

2) A boron concentration between 2000 and 2200 ppm,

3) A minimum solution temperature of 40°F, and

4) A maximum solution temperature of 50°F

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

ACTION:

a. With the Boric Acid Storage System inoperable, 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 1.6% $\Delta k/k$ 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 RWST 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 Transfer Pump Room temperature and the boric acid storage tank solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

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

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

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

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

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

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

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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) With four loops operating, 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, or

e) With three loops operating, the THERMAL POWER level is reduced to less than or equal to 50% of RATED THERMAL POWER within the next hour and within the following 4 hours the Neutron Flux High Trip Setpoint is reduced to less than or equal to 60% of RATED THERMAL POWER.

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

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

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

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

SURVEILLANCE REQUIREMENTS

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

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

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. With four loops operating, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours, or

3. With three loops operating, reduce THERMAL POWER to less than 32% of RATED THERMAL POWER within 8 hours.

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

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. With four loops operating, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours, or

3. With three loops operating, reduce THERMAL POWER to less than 32% 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 indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at least once per 18 months.

---

*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 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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 65% of RATED THERMAL POWER with the reactor coolant stop valves in the nonoperating loop closed.

SURVEILLANCE REQUIREMENTS

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

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

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

c. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be 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 Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

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

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

b. At 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 ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 3.1-2.

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 figures, or

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

SURVEILLANCE REQUIREMENTS

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

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

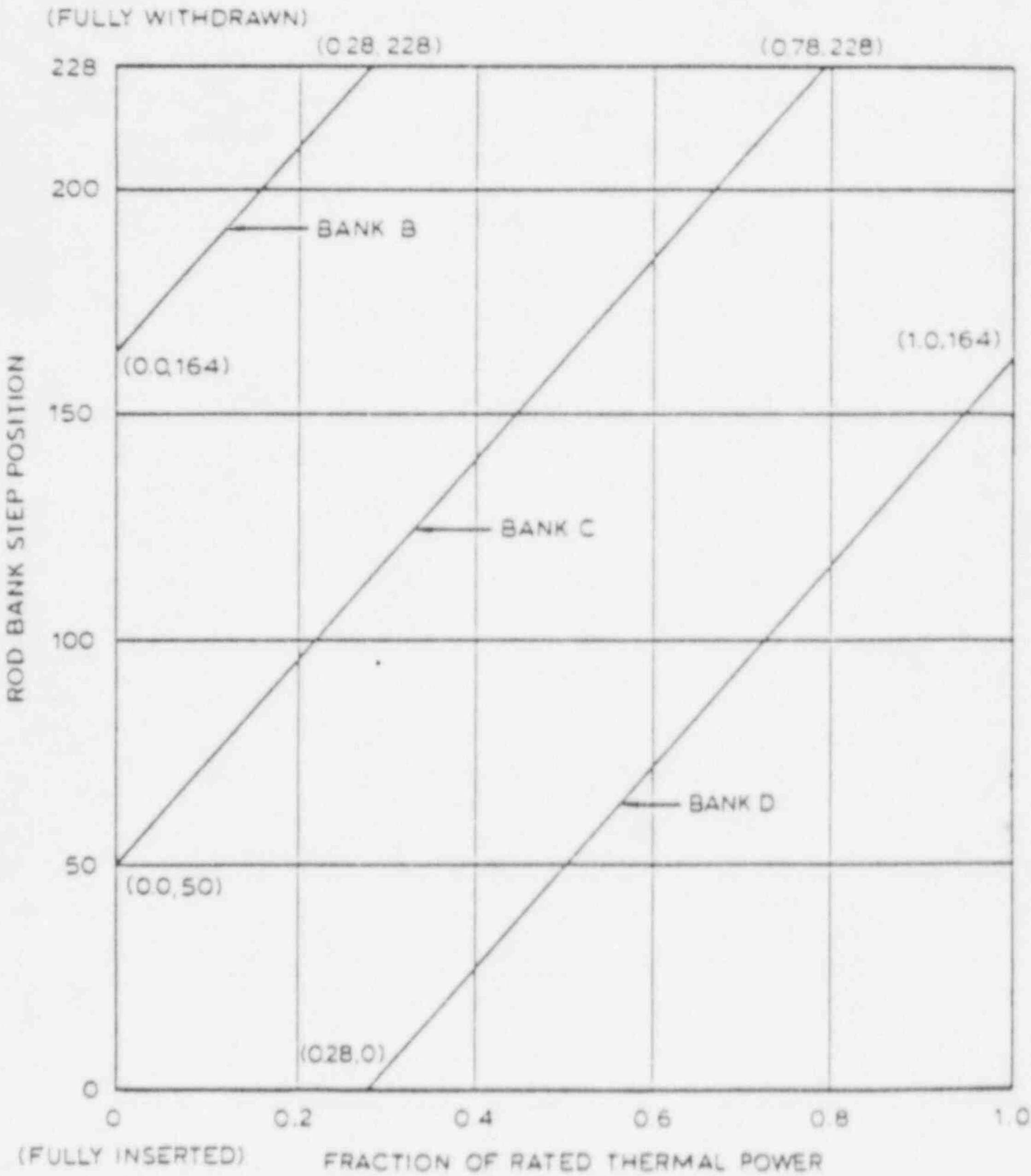


FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

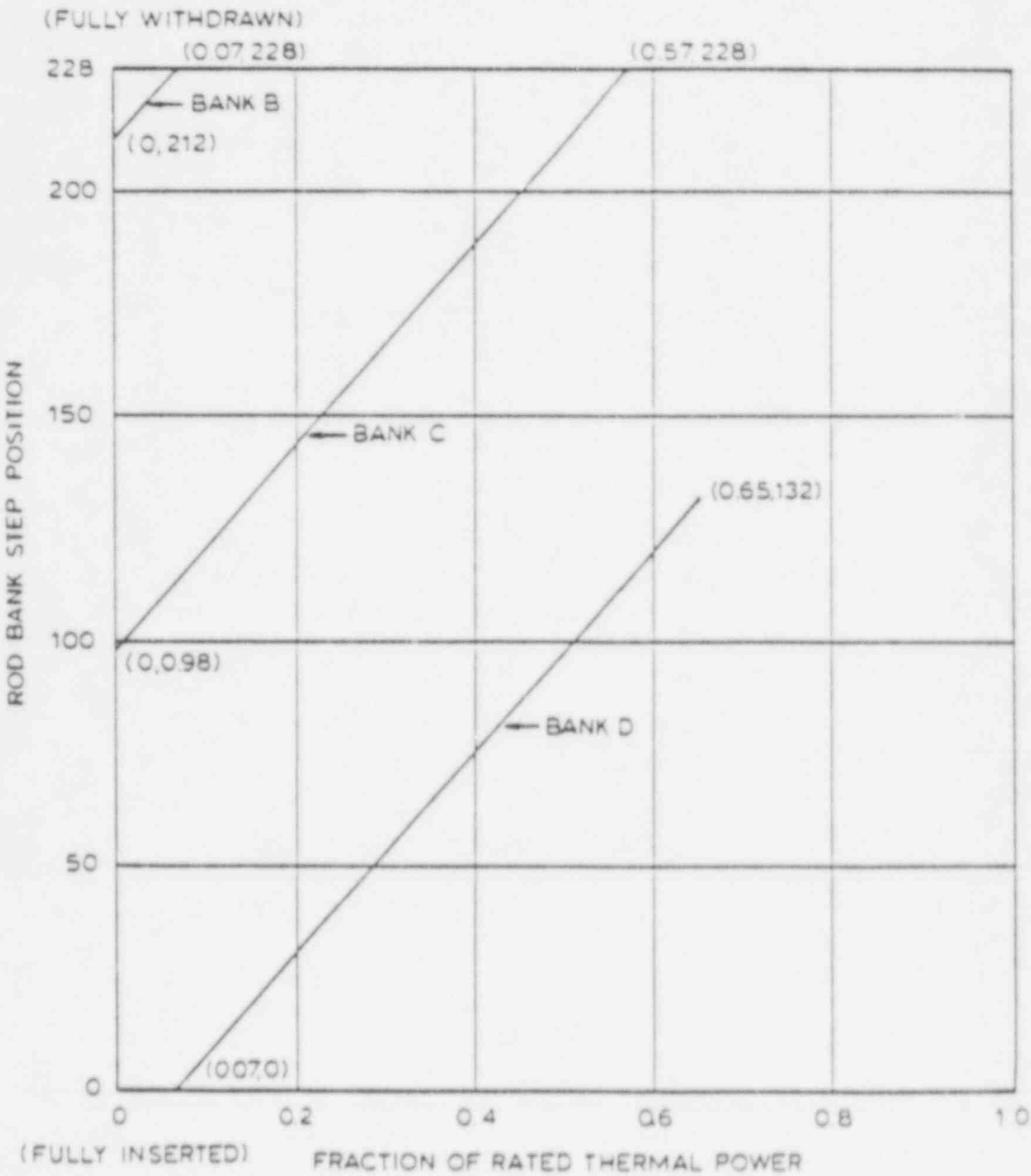


FIGURE 3.1-2

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
THREE LOOP OPERATION

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.1.1 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 core average accumulated burnup of less than or equal to 3000 MWD/MTU; and

b. + 3%, -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

The indicated AFD may deviate outside the above required target band 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 3.2-1a 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 time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1a 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

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

---

*See Special Test Exceptions Specification 3.10.2.

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

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

4.2.1.1.1 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.

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

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

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

4.2.1.1.3 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.

4.2.1.1.4 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 4.2.1.1.3 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.

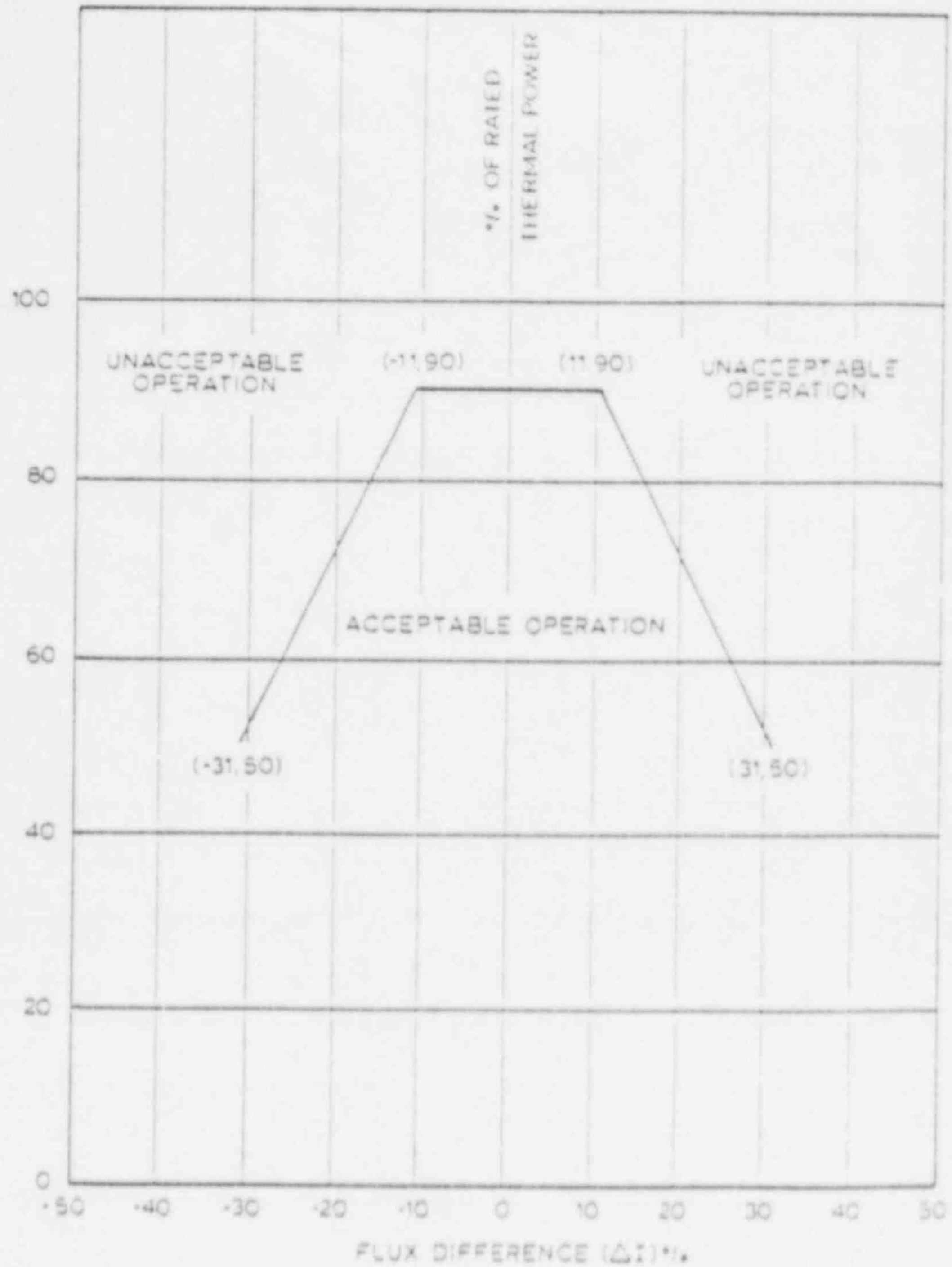


FIGURE 3.2-1a

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER
(FOUR LOOPS OPERATING)

POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE

THREE LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.1.2 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within ±5% of the target band (flux difference units) about the target flux difference.

The indicated AFD may deviate outside the above required target band at greater than or equal to 32% but less than 65% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1b 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 32% 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 65% 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 65% 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 time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1b and with THERMAL POWER less than 65% but equal to or greater than 32% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 32% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux - High** Setpoints to less than or equal to 37% of RATED THERMAL POWER within the next 4 hours.

*See Special Test Exceptions Specification 3.10.2.

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

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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 32% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 32% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.2.1 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.

4.2.1.2.2 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 32% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 32% of RATED THERMAL POWER.

4.2.1.2.3 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.

4.2.1.2.4 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 4.2.1.2.3 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.

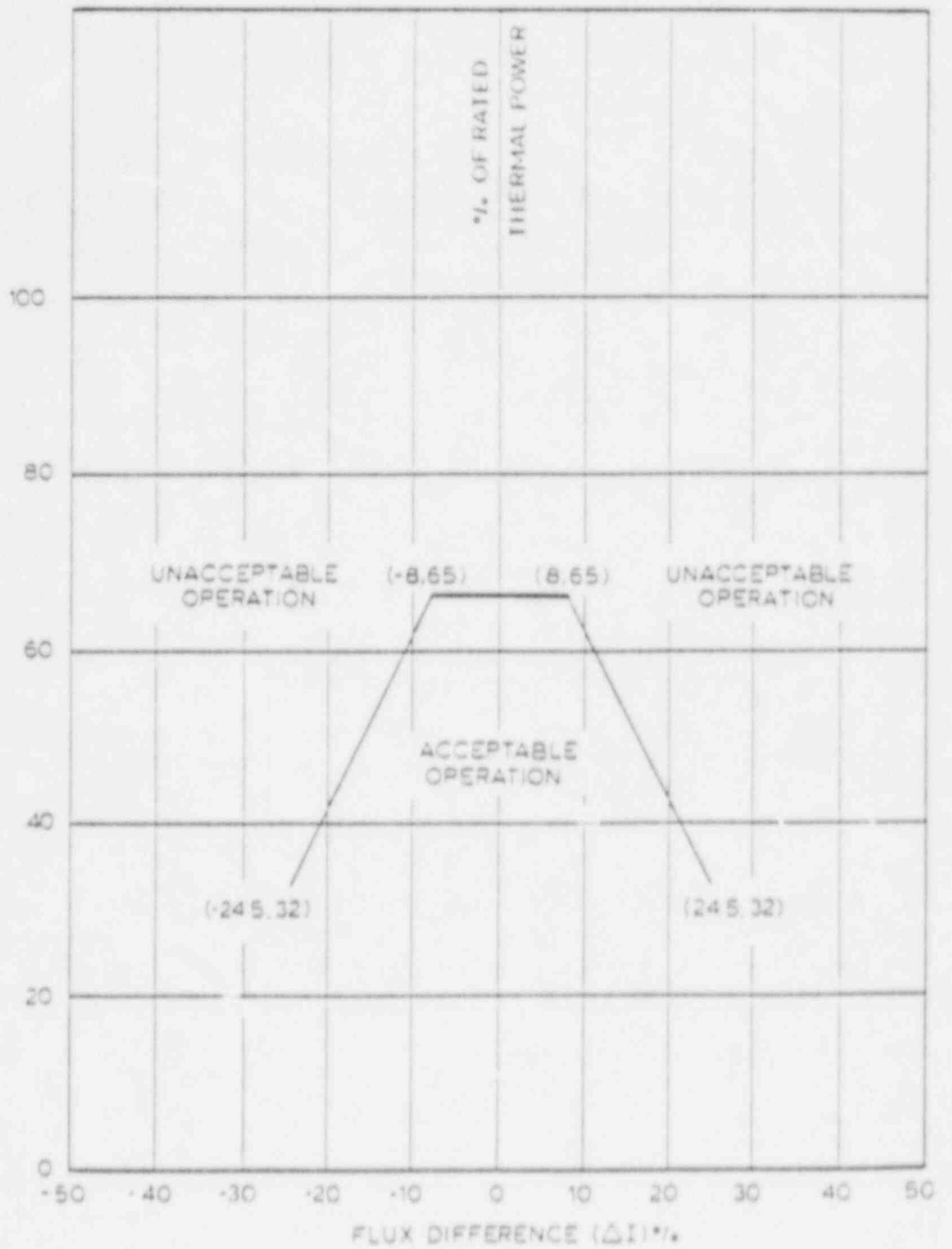


FIGURE 3.2-1b

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER
(THREE LOOPS OPERATING)

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

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.2.1 $F_Q(Z)$ shall be limited by the following relationships:

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

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

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

K(Z) = the function obtained from Figure 3.2-2a 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

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 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 4.2.2.1.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.1.2e. 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 Effective Full Power Days (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 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.1.2e., 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 18.1 ± 2%, 32.3 ± 2%, 46.6 ± 2%, 60.9 ± 2%, and 75.1 ± 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.

4.2.2.1.3 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.

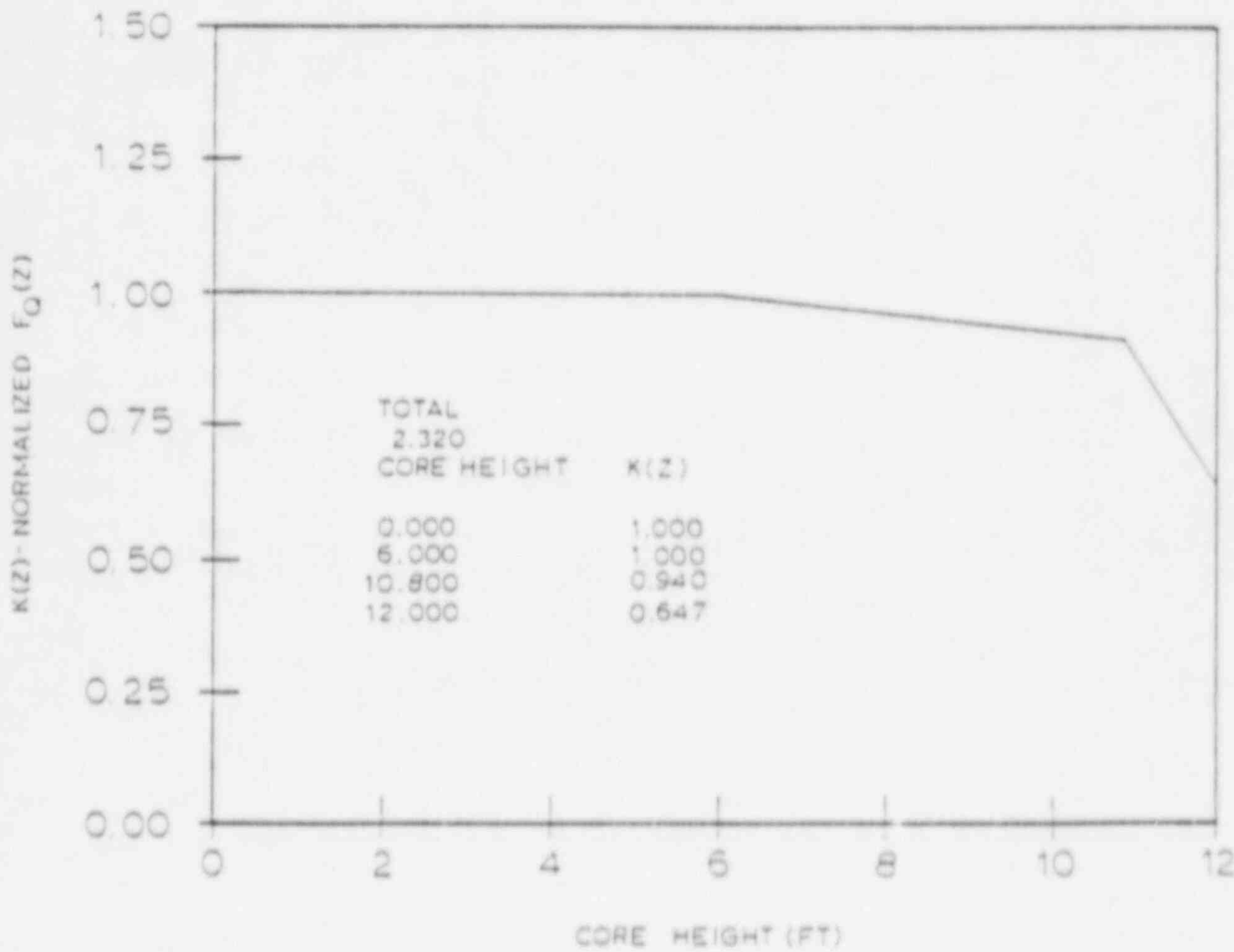


FIGURE 3.2-2a

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT
FOR FOUR LOOP OPERATION

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

THREE LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \leq \frac{1.69}{P} [K(Z)]$ for $P > 0.325$

$F_Q(Z) \leq (5.20) [K(Z)]$ for $P \leq 0.325$

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

K(Z) = the function obtained from Figure 3.2-2b 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. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

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

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 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 4.2.2.2.2b, above to:

1) The F_{xy} limits for 65% of RATED THERMAL POWER ($F_{xy}^{0.65\ RTP}$) for the appropriate measured core planes given in Specification 4.2.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^L = F_{xy}^{0.65\ RTP}\left[1+M_{F_{xy}}(1-P)\right],$$

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of $F_{xy}^{0.65\ RTP}$, $M_{F_{xy}}$ is the F_{xy} multiplier, 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 $F_{xy}^{0.65\ RTP}$ 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 $F_{xy}^{0.65\ RTP}$ 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 Effective Full Power Days (EFPD), whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued)

2) When the F_{xy}^{C} is less than or equal to the $F_{xy}^{0.65\ RTP}$ limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to $F_{xy}^{0.65\ RTP}$ and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for 65% of RATED THERMAL POWER ($F_{xy}^{0.65\ RTP}$) and the F_{xy} multiplier ($M_{F_{xy}}$) 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 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.2.2e., 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 18.1 ± 2%, 32.3 ± 2%, 46.6 ± 2%, 60.9 ± 2%, and 75.1 ± 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.

4.2.2.2.3 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.

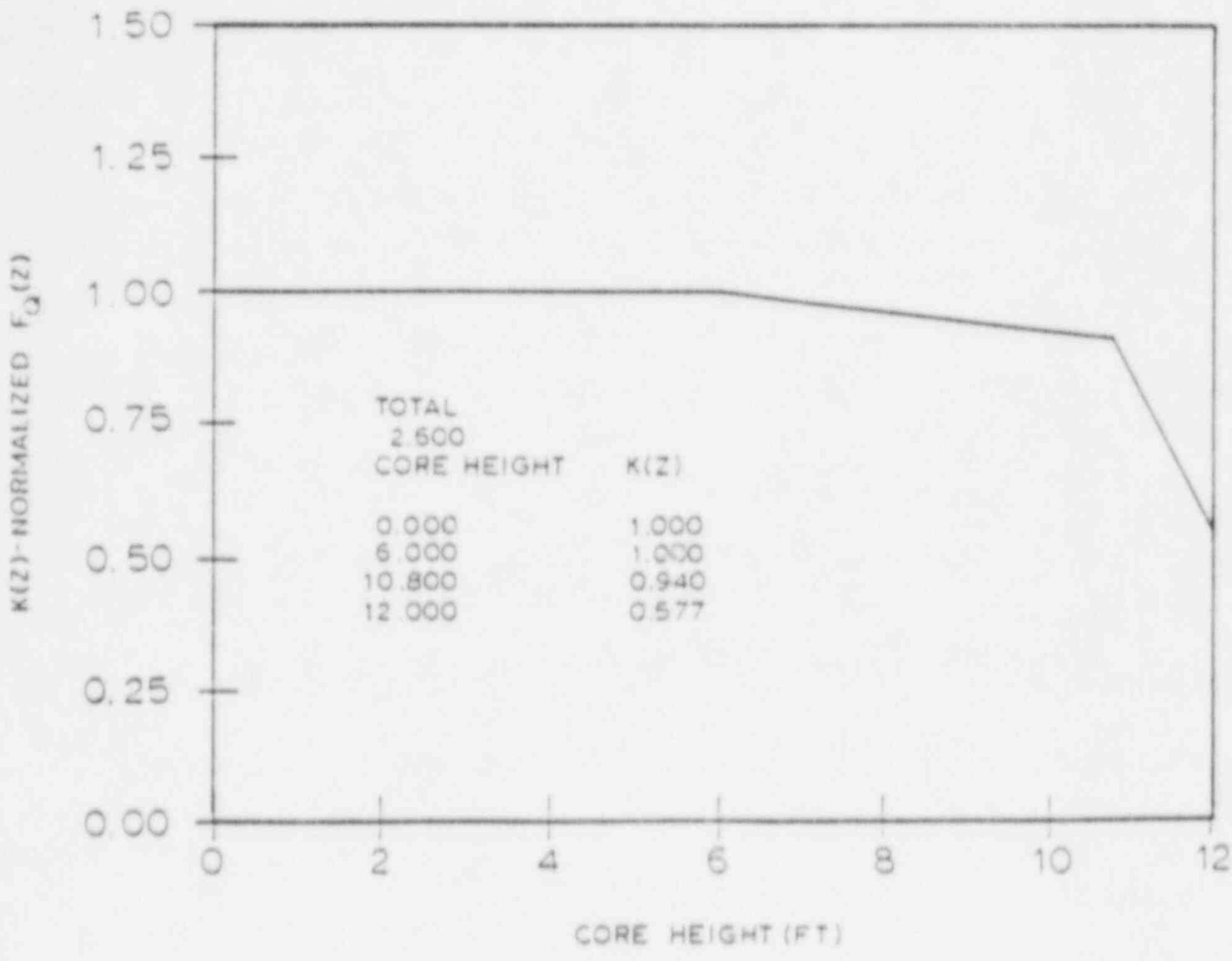


FIGURE 3.2-2b

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT
FOR THREE LOOP OPERATION

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

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 387,500 gpm, and

b. $F^N_{\Delta H} \leq 1.49\ [1.0 + 0.3\ (1.0 - P)]$

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement, and

3) The measured value of RCS total flow rate shall be used since uncertainties of 2.4% for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate and $F^N_{\Delta H}$ 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.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

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 $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation 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

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the acceptable range:

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.

4.2.3.1.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.1.2, is assumed to exist.

4.2.3.1.4 The RCS 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.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.1.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.3.1.6 If the feedwater venturis are not inspected at least once per 18 months, an additional 0.1% will be added to the total RCS flow measurement uncertainty.

POWER DISTRIBUTION LIMITS

RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

THREE LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.3.2 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 307,050 gpm, and

b. $F^N_{\Delta H} \leq 1.351 [1.0 + 0.43 (1.0 - P)]$

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.2b. takes into consideration a measurement uncertainty of 4% for incore measurement, and

3) The measured value of RCS total flow rate shall be used since uncertainties of 2.76% for flow measurement have been included in Specification 3.2.3.2a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 32% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 37% of RATED THERMAL POWER within the next 4 hours.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

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 $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 32% of RATED THERMAL POWER, and

2. A nominal 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2.2 RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the acceptable range at least once per 31 Effective Full Power Days.

4.2.3.2.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.2.2, is assumed to exist.

4.2.3.2.4 The RCS 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.

4.2.3.2.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.3.2.6 If the feedwater venturis are not inspected at least once per 18 months, an additional 0.1% will be added to the total RCS flow measurement uncertainty.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

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

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

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

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

   2. Within 2 hours either:

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

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 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.

---

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by 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 its limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: Four Loops in Operation	LIMITS: Three Loops in Operation & Loop Stop Valves Closed
Indicated Reactor Coolant System T_{avg}	$\leq$ 589.2°F	$\leq$ 581.7°F
Indicated Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*

*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*	11
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	0	1	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	11
7. Overtemperature ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	3	2	2	1, 2	6#
8. Overpower ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	3	2	2	1, 2	6#
9. Pressurizer Pressure--Low	4	2	3	1**	6# (1)
10. Pressurizer Pressure--High	4	2	3	1, 2	6# (1)
11. Pressurizer Water Level--High	3	2	2	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
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop in each operating loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
b. Two Loops (Above P-7 and below P-8)	3/loop in each operating loop	2/loop in two operating loops	2/loop each operating loop	1	6#
13. Steam Generator Water Level--Low-Low	4/stm. gen. in each operating stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6# (1)
14. Low Shaft Speed--Reactor Coolant Pumps					
a. Four loop operation	4-1/pump	2	3	1**	6#
b. Three loop operation	3-1/pump	2	2	1**	6#
15. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1***	12#
b. Turbine Stop Valve Closure	4	4	4	1***	6#
16. Safety Injection Input from ESF	2	1	2	1, 2	10
17. 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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
17. Reactor Trip System Interlocks (Continued)					
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1,2	8
18. Reactor Trip Breakers	2	1	2	1, 2	10, 13
	2	1	2	3*, 4*, 5*	11
19. Automatic Trip and Interlock Logic	2	1	2	1, 2	10
	2	1	2	3*, 4*, 5*	11
20. Three Loop Operation Bypass Circuitry	8 (1 switch per loop in each train)	2 (From different loop switches in bypass)	8	1, 2	1
21. Reactor Trip Bypass Breakers	2	1	2	1, 2	10
	2	1	2	3*, 4*, 5*	11

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

**Above the P-7 (At Power) Setpoint.

***Above the P-9 (Reactor Trip/Turbine Trip Interlock) Setpoint.

#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.

(1) The applicable MODES and ACTION statements for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

ACTION STATEMENTS

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

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

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 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 for four loop operation or 50% of RATED THERMAL POWER for three loop operation and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER for four loop operation or 60% of RATED THERMAL POWER for three loop operation within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

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

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 System breakers, suspend all operations involving positive reactivity changes and verify Valve 3CHS-V305 is 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 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - (Not used)

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 10 - 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 11 - 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 System breakers within the next hour.

ACTION 12 - 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 6 hours, and

b. When the Minimum Channels OPERABLE requirement is met, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of the Turbine Control Valves.

ACTION 13 - With one of the diverse trip features (undervoltage or shunt trip attachments) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-2

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. Reactor Coolant Flow--Low	
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 seconds
14. Low Shaft Speed-Reactor Coolant Pumps	$\leq$ 0.6 second**
15. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
16. Safety Injection Input from ESF	N.A.
17. Reactor Trip System Interlocks	N.A.
18. Reactor Trip Breakers	N.A.
19. Automatic Trip and Interlock Logic	N.A.
20. Three Loop Operation Bypass Circuitry	N.A.
21. Reactor Trip Bypass Breakers	N.A.

**Speed sensors are exempt from response time testing. Response time of the speed signal portion of the channel shall be measured from detector output or first electronic component in the channel.

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(14)	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)	Q(17)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1), Q(9,17)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	S	R(12)	Q(17)	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q(17)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(17,18)	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	Q(17,18)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	Q(17)	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	S	R	Q(17)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	Q(17,18)	N.A.	N.A.	1, 2
14. Low Shaft Speed - Reactor Coolant Pumps	N.A.	R(13)	Q(17)	N.A.	N.A.	1
15. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1
e. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	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 Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
20. Three Loop Operation Bypass Circuitry	N.A.	N.A.	N.A.	R	N.A.	1, 2
21. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	M(15) R(16)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

(1) If not performed in previous 31 days.

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

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

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

(5) Detector plateau curves shall be obtained, and 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) (Not used)

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of less than or equal to 5 times background.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

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

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

(13) Reactor Coolant Pump Shaft Speed Sensor may be excluded from CHANNEL CALIBRATION.

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

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

(16) Automatic undervoltage trip.

(17) Each channel shall be tested at least every 92 days on a STAGGERED TEST BASIS.

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

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 Value 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 ESFAS 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, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure--Low	4	2	3	1, 2, 3#	20*
e. Steam Line Pressure--Low	3/steam line in each operating loop	2/steam line in any operating loop	2/steam line in each operating loop	1, 2, 3#	15*
2. Containment Spray (CDA)					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	19

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 (CDA) (Continued)					
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-3	4	2	3	1, 2, 3	17
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	19
2) Automatic Actuation Logic 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					
1) Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	19
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

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)					
3) Containment Pressure--High-3	4	2	3	1, 2, 3	17
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	24
2) System	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Containment Pressure--High-2	3	2	2	1, 2, 3	15*
d. Steam Line Pressure--Low	3/steam line in each operating loop	2/steam line in any operating loop	2/steam line in each operating loop	1, 2, 3#	15*
e. Steam Line Pressure - Negative Rate--High	3/steam line in each operating loop	2/steam line in any operating loop	2/steam line in each operating loop	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
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	25
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen. in each operating loop	2/stm. gen. in any operating loop	3/stm. gen. in each operating loop	1, 2	20*
c. Safety Injection Actuation Logic	2	1	2	1, 2	22
d. T_{ave} Low Coincident with P-4					
1) Four Loops Operating	1 T_{ave}/loop	1 T_{ave} in any two loops	1 T_{ave} in any three loops	1, 2	20
2) Three Loops Operating	1 T_{ave} operating loop	1 T_{ave} in any two operating loops	1 T_{ave} in any two operating loops	1, 2	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
6. Auxiliary Feedwater					
a. Manual Initiation	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
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	20*
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	20*
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	2	1	2	1, 2, 3	19

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps	See Item 2. above for all CDA functions and requirements.				
7. Control Building Isolation					
a. Manual Actuation	2	1	2	All	19
b. Manual Safety Injection Actuation	2	1	2	1, 2, 3, 4	19
c. Automatic Actuation Logic and Actuation Relays	2	1	2	All	14
d. Containment Pressure--High-1	3	2	2	1, 2, 3	15
e. Control Building Inlet Ventilation Radiation	2/intake	1	2/intake	All	18
f. Outside Chlorine High	1/train	1	1/train	All	18
8. Loss of Power					
a. 4 kV Bus Undervoltage-Loss of Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20*
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	2	2	1, 2, 3	23
d. Steam Generator Water Level--High-High (P-14)	4/stm. gen. in each operating loop	2/stm. gen. in any operating loop	3/stm. gen. in each operating loop	1, 2, 3	21
10. Emergency Generator Load Sequencer	2	1	2	1, 2, 3, 4	14

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

*The provisions of Specification 3.0.4 are not applicable.

#The Steamline Isolation Logic and Safety Injection Logic for this trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

**The Safety Injection Logic for this trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) Setpoint.

***The channel(s) associated with the protective functions derived from the out of service reactor coolant loop shall be placed in the tripped mode.

****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.

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 a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - 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 18 - With less than the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 19 - 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 20 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. 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.

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

ACTION 22 - 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 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

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

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, 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, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High 1	3.3	1.01	1.75	≤ 3.0 psig	≤ 3.8 psig
d. Pressurizer Pressure--Low	16.5	13.67	3.3	≥ 1877.3 psig	≥ 1870.2 psig
e. Steam Line Pressure--Low	17.7	15.31	2.2	≥ 658.6 psig*	≥ 644.9 psig*
2. Containment Spray (CDA)					
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-3	3.3	1.01	1.75	≤ 8.0 psig	≤ 8.8 psig
3. Containment Isolation					
a. Phase "A" Isolation					
1) 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
3. Containment Isolation (Continued)					
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 Trip Setpoints and Allowable Values.				
b. Phase "B" 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) Containment Pressure--High-3	3.3	1.01	1.75	$\leq$ 8.0 psig	$\leq$ 8.8 psig
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-2	3.3	1.01	1.75	$\leq$ 3.0 psig	$\leq$ 3.8 psig
d. Steam Line Pressure--Low	17.7	15.31	2.2	$\geq$ 658.6 psig*	$\geq$ 644.9 psig*
e. Steam Line Pressure - Negative Rate--High	5.0	0.5	0	$\leq$ 100 psi/s**	$\leq$ 122.7 psi/s**

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
5. Turbine Trip and 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)	3.7	2.33	1.75	$\leq$ 82.0% of narrow range instrument span.	$\leq$ 82.8% of narrow range instrument span.
c. Safety Injection Actuation Logic	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
d. T_{ave} Low Coincident with Reactor Trip (P-4)					
1) Four Loops Operating	N.A.	N.A.	N.A.	$\geq$ 564°F	$\geq$ 562°F
2) Three Loops Operating	N.A.	N.A.	N.A.	$\geq$ 564°F	$\geq$ 562°F
6. 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.
c. Steam Generator Water Level--Low-Low					
1) Start Motor-Driven Pumps	20.5	18.98	1.75	$\geq$ 23.5% of narrow range instrument span.	$\geq$ 22.6% of narrow range instrument span.

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. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pumps	20.5	18.98	1.75	$\geq$ 23.5% of narrow range instrument span.	$\geq$ 22.6% of narrow range instrument span.
d. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps	N.A.	N.A.	N.A.	$\geq$ 2800V	$\geq$ 2720V
f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps	See Item 2. above for all CDA Trip Setpoints and Allowable Values.				
7. Control Building Isolation					
a. Manual Actuation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	N.A.	N.A.
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High 1	3.3	1.01	1.75	$\leq$ 3.0 psig	$\leq$ 3.8 psig
e. Control Building Inlet Ventilation Radiation	N.A.	N.A.	N.A.	$\leq 1.5\times10^{-5}\mu c/cc$	$\leq 1.5\times10^{-5}\mu c/cc$
f. Outside Chlorine High	N.A.	N.A.	N.A.	$\leq$ 5 ppm	$\leq$ 5 ppm

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. Loss of Power					
a. 4 kV Bus Undervoltage (Loss of Voltage)	N.A.	N.A.	N.A.	≥ 2800 volts with a ≤ 2 second time delay.	≥ 2720 volts with a ≤ 2 second time delay.
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	N.A.	N.A.	≥ 3710 volts with a ≤ 8 second time delay with ESF actuation or ≤ 300 second time delay without ESF actuation.	≥ 3706 volts with a ≤ 8 second time delay with ESF actuation or ≤ 300 second time delay without ESF actuation.
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1985 psig	≤ 1995 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	≥ 553°F	≥ 549.7°F
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				
10. Emergency Generator Load Sequencer	N.A.	N.A.	N.A.	N.A.	N.A.

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 less 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. Steam Line Isolation	N.A.
f. Feedwater Isolation	N.A.
g. Auxiliary Feedwater	N.A.
h. Service Water	N.A.
i. Control Building Isolation	N.A.
j. Reactor Trip	N.A.
k. Start Diesel Generator	N.A.
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 6.8^{(3)}$
3) Phase "A" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
4) Auxiliary Feedwater	≤ 60
5) Service Water	$\leq 90^{(1)}$
6) Start Diesel Generator	≤ 12
b. Control Building Isolation	≤ 5
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 6.8^{(3)}$
3) Phase "A" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
4) Auxiliary Feedwater	≤ 60
5) Service Water	$\leq 90^{(1)}$
6) Start Diesel Generators	≤ 12

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	$\leq 12^{(5)}/22^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 6.8^{(3)}$
3) Phase "A" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
4) Auxiliary Feedwater	≤ 60
5) Service Water	$\leq 90^{(1)}$
6) Start Diesel Generators	≤ 12
b. Steam Line Isolation	$\leq 6.8^{(3)}$
5. Containment Pressure--High-3	
a. Quench Spray	$\leq 32^{(2)}/42^{(1)}$
b. Phase "B" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
c. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
d. Service Water	$\leq 90^{(1)}$
6. Containment Pressure--High-2	
a. Steam Line Isolation	$\leq 6.8^{(3)}$
7. Steam Line Pressure - Negative Rate--High	
a. Steam Line Isolation	$\leq 6.8^{(3)}$
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	$\leq 6.8^{(3)}$
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60
10. Loss-of-Offsite Power	
a. Motor-Driven Auxiliary Feedwater Pump	≤ 60

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
11. Loss of Power	
a. 4 kV Bus Undervoltage (Loss of Voltage)	$\leq$ 12
b. 4 kV Emergency Bus Undervoltage (Grid Degraded Voltage)	$\leq 18^{(7)}/310^{(8)}$
12. T_{ave} Low Coincident With Reactor Trip (P-4)	
a. Feedwater Isolation	$\leq 6.8^{(3)}$
13. Control Building Inlet Ventilation Radiation	
a. Control Building Isolation	$\leq$ 3.7
14. Outside Chlorine High	
a. Control Building Isolation	$\leq$ 7

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air-operated valves.

(4) Diesel generator starting and sequence loading delay included. RHR pumps not included.

(5) Diesel generator starting and sequence loading delays not included. RHR pumps not included.

(6) Time required to close valves as indicated in Table 3.6-2.

(7) With an ESF signal present.

(8) Without an ESF signal present.

TABLE 4.3-2

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
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1	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
c. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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
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								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
4. Steam Line Isolation								
a. Manual Initiation								
1) Individual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
2) System	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

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
4. Steam Line Isolation (Continued)								
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-2	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. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection Actuation Logic	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
d. T_{ave} Low Coincident with Reactor Trip (P-4)	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2

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
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation 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	N.A.	N.A.	N.A	1, 2, 3
f. Containment Depressurization Actuation (CDA)	See Item 2. above for all CDA Surveillance Requirements.							
7. Control Building Isolation								
a. Manual Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
d. Containment Pressure--High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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
7. Control Building Isolation (Continued)								
e. Control Building Inlet Ventilation Radiation	S	R	M	N.A.	N.A.	N.A.	N.A.	All
f. Outside Chlorine High	S	R	M	N.A.	N.A.	N.A.	N.A.	All
8. Loss of Power								
a. 4 kV Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
9. 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. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level, P-14	S	R	M	N.A.	M(1)	M(1)	Q	1, 2, 3
10. Emergency Generator Load Sequencer	N.A.	N.A.	N.A.	N.A.	Q(1, 2)	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

(2) This surveillance may be performed continuously by the emergency generator load sequencer auto test system as long as the EGLS auto test system is demonstrated operable by the performance of an ACTUATION LOGIC TEST at least once per 92 days.

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 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 Area Purge and Exhaust Isolation	1	2	All	$\leq$ 1 R/h	26
b. RCS Leakage Detection					
1) Particulate Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2) Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2. Fuel Storage Pool Areas					
a. Criticality-Radiation Level	1	2	*	$\leq$ 15 mR/h	28
3. Control Room					
a. Air Intake-Radiation Level	1/intake	2/intake	All	$\leq 1.5\times10^{-5}$ μC/cc	27

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* With fuel in the fuel storage pool areas.

ACTION STATEMENTS

ACTION 26 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 27 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 28 - 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 storage pool areas.

ACTION 29 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

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 Area Purge and Exhaust Isolation	S	R	M	All
b. RCS Leakage Detection				
1) Particulate Radioactivity	S	R	M	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	M	1, 2, 3, 4
2. Fuel Storage Pool Areas				
a. Criticality-Radiation Level	S	R	M	*
3. Control Room				
a. Air Intake Radiation Level	S	R	M	All

TABLE NOTATIONS

* With fuel in the fuel storage pool area.

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 normalizing each detector output when required for:

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

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ 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 at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above required seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 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. NBE20A Containment Mat. (-24'3")	± 1g (5v/g)	1
b. NBE20B Containment Wall (40'6")	± 1g (5v/g)	1
c. NBE21 Emer. Generator Enclosure Located on Mat in Diesel Fuel Oil Vault (4'6")	± 1g (5v/g)	1
d. NBE22 Aux. Bldg. F-Line Wall Near The Charging Pumps Cooling Surge Tank (46'6")	± 1g (5v/g)	1
2. Triaxial Peak Accelerographs		
a. P/A1 Containment Safety Injection Accum. Tank (-4'7")	± 2g	1
b. P/A2 Safety Injection Accum. Disch. Line (-22'10")	± 1g	1
c. P/A3 Aux. Bldg. Charging Pumps Cooling Surge Tank (46'6")	± 1g	1
3a. Triaxial Seismic Trigger		
Horizontal (Control Room)	.01g	1*
Vertical (Control Room)	.006g	1*
3b. Triaxial Seismic Switch		
Horizontal (Control Room)	.09g	1**
Vertical (Control Room)	.06g	1**
4. Triaxial Response-Spectrum Recorders		
a. RSA-50 Spectrum Analyzer (Control Room)	1-32 Hz Peak Acceleration in Gs (Max of 1g)	1*
b. Self-Contained Recorder Steam Generator Support (51'4")	0-30 Hz at ± 2g	1

*With reactor control room indicator. This unit is activated by signals from the NBE20A Triaxial Accelerograph.

**This unit is activated by signals from the NBE20A Triaxial Accelerograph and is connected to an annunciator in the reactor control room.

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. NBE20A Containment Mat (-24'3")	M	R	SA
b. NBE20B Containment Wall (40'6")	M	R	SA
c. NBE21 Emer. Generator Enclosure Located on Mat in Diesel Fuel Oil Vault (4'6")	M	R	SA
d. NBE22 Aux. Bldg. F-Line Wall Near The Charging Pumps Cooling Surge Tank (46'6")	M	R	SA
2. Triaxial Peak Accelerographs			
a. P/A1 Containment Safety Injection Accum. Tank (-4'7")	N.A.	R	N.A.
b. P/A2 Safety Injection Accum. Disch. Line (-22'10")	N.A.	R	N.A.
c. P/A3 Aux. Bldg. Charging Pumps Cooling Surge Tank (46'6")	N.A.	R	N.A.
3a. Triaxial Seismic Trigger			
Horizontal (Control Room)	M	R	SA
Vertical (Control Room)	M	R	SA
3b. Triaxial Seismic Switch			
Horizontal (Control Room)	M	R	SA
Vertical (Control Room)	M	R	SA
4. Triaxial Response-Spectrum Recorders			
a. RSA-50 Spectrum Analyzer (Control Room)	M	R	SA
b. Self-Contained Recorder Steam Generator Support (51'4")	N.A.	R	N.A.

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 at the frequencies shown in Table 4.3-5.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT		LOCATION	MINIMUM OPERABLE
1.	Wind Speed		
a.	WS-33	Nominal Elev. 33'	1
b.	WS-142	Nominal Elev. 142'	1
c.	WS-374	Nominal Elev. 374'	1
2.	Wind Direction		
a.	WD-33	Nominal Elev. 33'	1
b.	WD-142	Nominal Elev. 142'	1
c.	WD-374	Nominal Elev. 374'	1
3.	Air Temperature - ΔT*		
a.	DT-142	Nominal Elev. 142'	1
b.	DT-374	Nominal Elev. 374'	1

*Group reference is 33'. ΔT is the measured difference between the temperature at 33' and the temperature at elevations 142' and 374', respectively.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 33'	D	SA
b. Nominal Elev. 142'	D	SA
c. Nominal Elev. 374'	D	SA
2. Wind Direction		
a. Nominal Elev. 33'	D	SA
b. Nominal Elev. 142'	D	SA
c. Nominal Elev. 374'	D	SA
3. Air Temperature - ΔT		
a. Nominal Elev. 33' - 142'	D	SA
b. Nominal Elev. 33' - 374'	D	SA

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown Instrumentation transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

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 SHUTDOWN within the next 12 hours.

b. With one or more Remote Shutdown Instrumentation transfer switches, power, or control circuits inoperable, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 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.

4.3.3.5.2 Each Remote Shutdown Instrumentation transfer switch, power and control circuit including the actuated components, shall be demonstrated OPERABLE at least once per 18 months.

TABLE 3.3-9

REMOTE SHUTDOWN INSTRUMENTATION

INSTRUMENT	TOTAL NO. READOUT LOCATION	MINIMUM OF CHANNELS	CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	Reactor Trip Switchgear	1/trip breaker	1/trip breaker
2. Pressurizer Pressure	Aux. Shutdown Panel	2	1
3. Pressurizer Level	Aux. Shutdown Panel	2	1
4. Steam Generator Pressure	Aux. Shutdown Panel	2/steam generator	1/steam generator
5. Steam Generator Water Level	Aux. Shutdown Panel	2/steam generator	1/steam generator
6. Auxiliary Feedwater Flow Rate	Aux. Shutdown Panel	1/steam generator	1/steam generator
7. Loop Hot Leg Temperature	Aux. Shutdown Panel	1/loop	1/loop
8. Loop Cold Leg Temperature	Aux. Shutdown Panel	1/loop	1/loop
9. Reactor Coolant System Pressure (Wide Range)	Aux. Shutdown Panel	2	1
10. DWST Level	Aux. Shutdown Panel	2	1
11. RWST Level	Aux. Shutdown Panel	2	1
12. Containment Pressure	Aux. Shutdown Panel	2	1
13. Emergency Bus Voltmeters	Aux. Shutdown Panel	1/train	1/train
14. Source Range Count Rate	Aux. Shutdown Panel	2	1
15. Intermediate Range Flux	Aux. Shutdown Panel	2	1
16. Boric Acid Tank Level	Aux. Shutdown Panel	2/tank	1/tank

TRANSFER SWITCHES	SWITCH LOCATION
1. Auxiliary Feedwater Isolation FWA*MOV35A	Transfer Switch Panel
2. Auxiliary Feedwater Isolation FWA*MOV35B	Transfer Switch Panel
3. Auxiliary Feedwater Isolation FWA*MOV35C	Transfer Switch Panel
4. Auxiliary Feedwater Isolation FWA*MOV35D	Transfer Switch Panel
5. Auxiliary Feedwater Pump Ah. Suction FWA*AOV23A	Transfer Switch Panel
6. Auxiliary Feedwater Pump Ah. Suction FWA*AOV23B	Transfer Switch Panel

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

TRANSFER SWITCHES	SWITCH LOCATION
7. Turbine Driven Pump Steam Supply MSS*AOV31A	Transfer Switch Panel
8. Turbine Driven Pump Steam Supply MSS*AOV31B	Transfer Switch Panel
9. Turbine Driven Pump Steam Supply MSS*AOV31D	Transfer Switch Panel
10. Reactor Vessel Head Vent Isolation RCS*SV8095A	Transfer Switch Panel
11. Reactor Vessel Head Vent Isolation RCS*SV8095B	Transfer Switch Panel
12. Reactor Vessel Head Vent Isolation RCS*SV8096A	Transfer Switch Panel
13. Reactor Vessel Head Vent Isolation RCS*SV8096B	Transfer Switch Panel
14. Reactor Vessel to Excess Letdown RCS*MV8098	Transfer Switch Panel
15. Pressurizer Level Control RCS*LCV459	Transfer Switch Panel
16. Pressurizer Level Control RCS*LCV460	Transfer Switch Panel
17. Letdown Orifice Isolation CHS*AV8149A	Transfer Switch Panel
18. Letdown Orifice Isolation CHS*AV8149B	Transfer Switch Panel
19. Letdown Orifice Isolation CHS*AV8149C	Transfer Switch Panel
20. Volume Control Tank Outlet Isolation CHS*LCV112B	Transfer Switch Panel
21. Volume Control Tank Outlet Isolation CHS*LCV112C	Transfer Switch Panel
22. RWST to CHS Pump Suction CHS*LCV112D	Transfer Switch Panel
23. RWST to CHS Pump Suction CHS*LCV112E	Transfer Switch Panel
24. Charging to RCS Isolation CHS*AV8146	Transfer Switch Panel
25. Charging to RCS Isolation CHS*AV8147	Transfer Switch Panel
26. Boric Acid Gravity Feed CHS*MV8507A	Transfer Switch Panel
27. Boric Acid Gravity Feed CHS*MV8507B	Transfer Switch Panel

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

TRANSFER SWITCHES	SWITCH LOCATION
28. Charging Header Isolation Bypass CHS*MV8116	Transfer Switch Panel
29. Pressurizer Heater Backup RCS*H1A (Group A)	Transfer Switch Panel
30. Pressurizer Heater Backup RCS*H1B (Group B)	Transfer Switch Panel

CONTROL CIRCUITS	SWITCH LOCATION
1. Auxiliary Feedwater Flow Control FWA*HV31A	Auxiliary Shutdown Panel
2. Auxiliary Feedwater Flow Control FWA*HV31B	Auxiliary Shutdown Panel
3. Auxiliary Feedwater Flow Control FWA*HV31C	Auxiliary Shutdown Panel
4. Auxiliary Feedwater Flow Control FWA*HV31D	Auxiliary Shutdown Panel
5. Auxiliary Feedwater Flow Control FWA*HV32A	Auxiliary Shutdown Panel
6. Auxiliary Feedwater Flow Control FWA*HV32B	Auxiliary Shutdown Panel
7. Auxiliary Feedwater Flow Control FWA*HV32C	Auxiliary Shutdown Panel
8. Auxiliary Feedwater Flow Control FWA*HV32D	Auxiliary Shutdown Panel
9. Auxiliary Feedwater Flow Control FWA*HV36A	Auxiliary Shutdown Panel
10. Auxiliary Feedwater Flow Control FWA*HV36B	Auxiliary Shutdown Panel
11. Auxiliary Feedwater Flow Control FWA*HV36C	Auxiliary Shutdown Panel
12. Auxiliary Feedwater Flow Control FWA*HV36D	Auxiliary Shutdown Panel

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

CONTROL CIRCUITS	SWITCH LOCATION
13. Reactor Vessel to PRT Control RCS*HCV442A	Auxiliary Shutdown Panel
14. Reactor Vessel to PRT Control RCS*HCV442B	Auxiliary Shutdown Panel
15. Charging Header Flow Control CHS*HCV190A	Auxiliary Shutdown Panel
16. Charging Header Flow Control CHS*HCV190B	Auxiliary Shutdown Panel
17. Excess Letdown Flow Control CHS*HCV123	Auxiliary Shutdown Panel
18. Charging Flow Control CHS*FCV121	Auxiliary Shutdown Panel
19. Low Pressure Letdown Control CHS*PCV131	Auxiliary 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. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Pressure	M	R
5. Steam Generator Water Level	M	R
6. Auxiliary Feedwater Flow Rate	M	R
7. Loop Hot Leg Temperature	M	R
8. Loop Cold Leg Temperature	M	R
9. Reactor Coolant System Pressure (Wide Range)	M	R
10. DWST Level	M	R
11. RWST Level	M	R
12. Containment Pressure	M	R
13. Emergency Bus Voltmeters	M	R
14. Source Range Count Rate	M	R
15. Intermediate Range Amps	M	R
16. Boric Acid Tank Level	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 except the containment area high range radiation monitor, 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 at least HOT SHUTDOWN within the following 6 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels except the containment area-high range radiation monitor, 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 at least HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE channels for the containment area-high range radiation monitor less than required by either the total or 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 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		
a. Normal Range	2	1
b. Extended Range	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	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Demineralized Water Storage Tank Water Level	2	1
11. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator
12. Reactor Coolant System Subcooling Margin Monitor	2	1
13. Containment Water Level (Wide Range)	2	1
14. Core Exit Thermocouples	4/core quadrant	2/core quadrant
15. Containment Area - Purge and Exhaust Isolation	2	1

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
16. Containment Area - High Range Radiation Monitor	2	1
17. Reactor Vessel Water Level	2	1
18. Containment Hydrogen Monitor	2	1
19. Neutron Flux	2	1

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure		
a. Normal Range	M	R
b. Extended Range	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. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Demineralized Water Storage Tank Water Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. Containment Water Level (Wide Range)	M	R
14. Core Exit Thermocouples	M	R
15. Containment Area - Purge and Exhaust Isolation	M	R

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
16. Containment Area - High Range Radiation Monitor	M	R*
17. Reactor Vessel Water Level	M	R
18. Containment Hydrogen Monitor	M	R
19. Neutron Flux	M	R

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

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 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 the next 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 4.6.1.6).

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 4.6.1.6).

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a FIRE DETECTOR OPERATIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a FIRE DETECTOR OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 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

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS*		
	HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1. Containment**			
a. Elevation 24'6"	8/0		
b. RCP Cubicle D	4/0		
c. RCP Cubicle A	4/0		
d. RCP Cubicle C	4/0		
e. RCP Cubicle B	4/0		
f. Electrical Penetration Area, El. 24'6"			16/0
g. Outer Annulus, El. 3'8" and 24'6"			16/0
2. Auxiliary Building			
a. East MCC Rod Area			0/16
b. West MCC Rod Area			0/16
c. North Floor Area, El. 4'6"			14/0
d. RPCCW Pump Area, El. 24'6"			9/0
e. Charging Pump Area			3/0
f. General Area, El. 43'6"			13/0
g. General Area, El. 66'6"			17/0
h. East MCC Rod Area - CO_2			0/12
i. West MCC Rod Area - CO_2			0/12
3. ESF Building			
a. RSS Pump Area			4/0
b. RSS Pump Area			4/0
c. RHR HX Area (North)			8/0
d. RHR HX Area (South)			4/0
e. General Area, El. 4'6"			2/0
f. FWA Pump Area			2/0
g. QSS Pump Area			4/0
h. FWA Pump Area			4/0
i. FWA Pump Area			5/0
j. North HVAC Area			2/0
k. South HVAC Area			2/0
l. H_2 Recombiner Bldg.			5/0

*(x/y): x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of Fire Suppression Systems and early warning and notification) instruments.

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

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION		TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
4.	Control Building			
a.	Switchgear Room A			0/19
b.	Cable Tray A, El. 4'6"			0/19
c.	Battery Room A			8/0
d.	Switchgear Room B			0/20
e.	Cable Tray B, El. 4'6"			0/17
f.	Battery Room B			6/0
g	NE Cable Spreading Room			0/8
h.	SE Cable Spreading Room			0/11
i.	NW Cable Spreading Room			0/8
j.	SW Cable Spreading Room			0/11
k.	Computer Room Floor			2/0
l.	East Instrument Rack Room Floor			3/0
m.	West Instrument Rack Room Floor			5/0
n.	Computer Room	0/4		4/0
o.	East Instrument Rack Room			7/0
p.	West Instrument Rack Room	0/17		12/0
q.	Control Room	1/0		27/0
r.	HVAC Room			9/0
s.	Chiller Room			3/0
t.	Switchgear Room A - CO_2			0/16
u.	Switchgear Room B - CO_2			0/15
v.	Cable Spreading Room - CO_2			0/15
w.	Cable Spreading Room - CO_2			0/19
5.	Emergency Diesel Building			
a.	Diesel Generator A Area	14/0	4/0	1/0
b.	Diesel Generator B Area	14/0	4/0	1/0
c.	Fuel Oil Tank Vault A	0/3		2/0
d.	Fuel Oil Tank Vault B	0/3		2/0
6.	Intake Structure			
a.	Circ. Water Pump Area			6/0
b.	Service Water Pump Area A			4/0
c.	Service Water Pump Area B			4/0
7.	Service Building			
a.	North Cable Tunnel			0/6
b.	South Cable Tunnel			0/7
c.	North Cable Tunnel - CO_2			0/5
d.	South Cable Tunnel - CO_2			0/6

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS*		
	HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
8. Fuel Building			
a. General Area			4/0
b. Fuel Pool Cooling Pump Area			17/0

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 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

4.3.3.8 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 at least once per 31 days, and

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

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 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 in accordance with the methodology and parameters as described in the REMODCM.

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, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Exert best efforts to restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner. Releases need not be terminated after 30 days provided the specified actions are continued.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 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 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 Neutralization Sump Monitor-Condensate Polishing Facility	1	31
b. Turbine Building Floor Drains	1	32***
c. Liquid Waste Monitor	1	31
d. Regenerate Evaporator Monitor-Condensate Polishing Facility	1*	32
e. Steam Generator Blowdown Monitor	1	32
2. Flow Rate Measurement Devices-No Alarm Setpoint Requirements		
a. Waste Neutralization Sump Effluents	1	33
b. Turbine Building Floor Drains	**	N.A.
c. Liquid Waste Effluent Line	1	33
d. Regenerate Evaporator Effluent Line	1*	33
e. Steam Generator Blowdown Effluent Line	1	33
f. Dilution Water Flow	**	N.A.

*N.A. if the Condensate Polishing Facility Regenerate Evaporator is not in service.
**Flow will be determined by pump status.
***N.A. if the Turbine Building sump is less than MDA.

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and 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. The original release rate calculations and discharge line valving are independently verified by a second individual.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided best efforts are made to repair the instrument and that grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of no more than 3×10^{-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 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

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 Releases				
a. Waste Neutralization Sump Monitor-Condensate Polishing Facility	D	P	R(2)	Q(1)
b. Turbine Building Floor Drains	D	M	R(2)	Q(1)
c. Liquid Waste Monitor	D	P	R(2)	Q(1)
d. Regenerate Evaporator Monitor-Condensate Polishing Facility (5)	D	M	R(2)	Q(1)
e. Steam Generator Blowdown Monitor	D	M	R(2)	Q(1)
2. Flow Rate Measurement Devices				
a. Waste Neutralization Sump Effluents	D(3)	NA	R	Q
b. Turbine Building Floor Drains	D(4)	NA	NA	NA
c. Liquid Waste Effluent Line	D(3)	NA	R	Q
d. Regenerate Evaporator Effluent Line (5)	D(3)	NA	R	Q
e. Steam Generator Blowdown Effluent Line	D(3)	NA	R	Q
f. Dilution Water Flow	D(4)	NA	NA	NA

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 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 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.

(4) Pump status shall be checked daily for the purpose of determining flowrate.

(5) Surveillance is required only if the monitor is required to be OPERABLE by Table 3.3-12.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive 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 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined in accordance with the methodology and parameters in the REMODCM.

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, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With the number of OPERABLE radioactive gaseous effluent monitoring instrumentation channels less than the Minimum Channels OPERABLE, take the ACTION shown in Table 3.3-13. Exert best efforts to restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner. Releases need not be terminated after 30 days provided the specified actions are continued.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 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 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. Millstone Unit 3 Ventilation Vent Stack (Turbine Building)			
a. Noble Gas Activity Monitor - Providing Alarm	1	*	34
b. Iodine Sampler	1	*	35
c. Particulate Sampler	1	*	35
d. Stack Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36
2. Millstone Unit 1 Main Stack			
a. Noble Gas Activity Monitor-Providing Alarm	1	*	37
b. Iodine Sampler	1	*	35
c. Particulate Sampler	1	*	35
d. Stack Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. Engineered Safeguards Building Monitor			
a. Noble Gas Activity Monitor-Providing Alarm	1	*	34
b. Iodine Sampler	1	*	35
c. Particulate Sampler	1	*	35
d. Discharge Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36
4. Warehouse No. 5 Vent			
a. Noble Gas Monitor	1(1)	**	34
b. Iodine Sampler	1(1)	**	34
c. Particulate Sampler	1(1)	**	34

TABLE 3.3-13 (Continued)

TABLE NOTATIONS

*At all times.

**When the gross activity of the regenerated waste is greater than 1×10^{-4} microCuries/ml.

(1)This minimum channel requirement may be met with a portable continuous air monitor (Eberline PING-3 or equivalent).

ACTION STATEMENTS

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that samples are continuously collected with auxiliary sampling equipment for periods of seven (7) days and analyzed for principal gamma emitters with half lives greater than 8 days within 48 hours after the end of the sampling period.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that best efforts are made to repair the instrument and that the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channnels OPERABLE less than required by the Minimum Channels OPERABLE requirement, Millstone Unit 3 releases via the Millstone Unit 1 stack may continue provided that best efforts are made to repair the instrument and that grab samples are taken at least once per 12 hours and analyzed for gross radioactivity within 24 hours.

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. Millstone Unit 3 Ventilation Vent Stack (Turbine Building)					
a. Noble Gas Activity Monitor	D	M	R(1)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Stack Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
2. Millstone Unit 1 Main Stack					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Stack Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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
3. Engineered Safeguards Building Monitor					
a. Noble Gas Activity Monitor	D	M	R(1)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Discharge Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
4. Warehouse No. 5 Vent					
a. Noble Gas Monitor	D	N.A.	R(3)	N.A.	**
b. Iodine Sampler	D	N.A.	R(3)	N.A.	**
c. Particulate Sampler	D	N.A.	R(3)	N.A.	**

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times except when the vent path is isolated.

** When the gross activity of the regenerated waste is greater than 1×10^{-4} microCuries/ml.

(1) 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.

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

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

b. Circuit failure (not applicable to Unit 1 Stack Monitor), or

c. Instrument indicates a downscale failure.

(3) The CHANNEL CALIBRATION shall include the use of a known source whose strength is determined by a detector which has been calibrated to an NBS source. These sources shall be in a known, reproducible geometry.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

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

APPICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat 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 by cycling each of the following valves through at least one complete cycle from the running position:

   1) Four high pressure main stop valves,

   2) Four high pressure turbine control valves, and

   3) Six low pressure combined intermediate valves.

b. At least once per 31 days by direct observation of the movement of each of the above 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 and performing a visual and surface inspection of valve seats, disks, and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

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 Either:

a) All reactor coolant loops shall be in operation, or

b) Three reactor coolant loops shall be in operation with THERMAL POWER restricted to less than or equal to 65% of RATED THERMAL POWER.

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

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE 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 17% at least once per 12 hours.

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

---

*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.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,**

e. RHR Loop 1, and

f. RHR Loop 2.

APPLICABILITY: MODE 4.

ACTION:

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

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

---

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause 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 350°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), 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 17% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR 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 (RHR) loop shall be OPERABLE and in operation*, and either:

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

b. The secondary side water level of at least two steam generators shall be greater than 17%.

APPLICABILITY: MODE 5 with at least two reactor coolant loops filled***.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

---

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

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

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with less than two reactor coolant loops filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 The required RHR loops shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

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

---

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

**The RHR pump may be 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.

ISOLATED LOOP

LIMITING CONDITION FOR OPERATION

3.4.1.5 The RCS loop stop valves of an isolated loop shall be shut and the power removed from the valve operators.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied: either shut the loop stop valves and remove power from the valve operators within one hour, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.5 The RCS loop stop valves of an isolated loop shall be verified shut and power removed from the valve operators at least once per 31 days.

REACTOR COOLANT SYSTEM

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated with power removed from the associated RCS loop stop valve operators until:

a. The temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops,

b. The boron concentration of the isolated loop is greater than or equal to the boron concentration of the operating loops,

c. The isolated portion of the loop has been drained and is refilled, and

d. The reactor is subcritical by at least 1.6% $\Delta k/k$.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the requirements of the above specification not satisfied, do not open the isolated loop stop valves.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.2 The reactor shall be determined to be subcritical by at least 1.6% $\Delta k/k$ within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.3 Within 4 hours prior to opening the loop stop valves, the isolated loop shall be determined to:

a. Be drained and refilled, and

b. Have a boron concentration greater than or equal to the boron concentration of the operating loops.

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 2500 psia ± 1%.*

APPLICABILITY: MODE 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 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 Surveillance Requirements other than those required by Specification 4.0.5.

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

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 92% (1656 cubic feet), and at least two groups of pressurizer heaters supplied by emergency power each having a capacity of at least 175 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, 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 HOT SHUTDOWN within the following 6 hours.

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

c. With both PORV(s) 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 in HOT SHUTDOWN within the following 6 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 assoicated 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 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by 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 associated with an operating RCS loop shall be OPERABLE.

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

ACTION:

With one or more steam generators associated with an operating RCS loop 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) Primary-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 Specification 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

SURVEILLANCE REQUIREMENTS (Continued)

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	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

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. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. 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 Drain Sump Level or Pumped Capacity Monitoring System, and

c. 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 Radioactive 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 Radioactivity Monitoring Systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment Drain Sump Level and Pumped Capacity Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

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 not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System,

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

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2250 ± 20 psia, and

f. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2250 ± 20 psia 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.

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 or particulate) radioactivity monitor at least once per 12 hours;

b. Monitoring the containment drain sump 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 2250 ± 20 psia at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

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

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

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

a. 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,

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

e. As outlined in the ASME Code, Section XI, paragraph IWV-3427(b).

The provisions of Specifica'ion 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
3-SIL-V15	SI Tank 1A Discharge Isolation Valve
3-SIL-V17	SI Tank 1B Discharge Isolation Valve
3-SIL-V19	SI Tank 1C Discharge Isolation Valve
3-SIL-V21	SI Tank 1D Discharge Isolation Valve
3-SIL-V26	RHR/SI to RCS Loop 2, Hot Leg
3-SIL-V27	SIH to RCS Loop 2, Hot Leg
3-SIL-V28	RHR/SI to RCS Loop 4, Hot Leg
3-SIL-V29	SIH to RCS Loop 4, Hot Leg
3-SIL-V984	RHR/SI to RCS Loop 4, Cold Leg
3-SIL-V985	RHR/SI to RCS Loop 3, Cold Leg
3-SIL-V986	RHR/SI to RCS Loop 2, Cold Leg
3-SIL-V987	RHR/SI to RCS Loop 1, Cold Leg
3-SIH-V5	SIH to RCS Cold Legs
3-SIH-V110	SIH to RCS Loop 1, Hot Leg
3-SIH-V112	SIH to RCS Loop 3, Hot Leg
3-RCS-V26	SIH to RCS Loop 1, Hot Leg
3-RCS-V29	SIH to RCS Loop 1, Cold Leg
3-RCS-V30	SIL to RCS Loop 1, Cold Leg
3-RCS-V69	RHR/SI to RCS Loop 2, Hot Leg
3-RCS-V70	SIH to RCS Loop 2, Cold Leg
3-RCS-V71	SIL to RCS Loop 2, Cold Leg
3-RCS-V102	SIH to RCS Loop 3, Hot Leg
3-RCS-V106	SIH to RCS Loop 3, Cold Leg
3-RCS-V107	SIL to RCS Loop 3, Cold Leg
3-RCS-V142	RHR/SI to RCS Loop 4, Hot Leg
3-RCS-V145	SIH to RCS Loop 4, Cold Leg
3-RCS-V146	SIL to RCS Loop 4, Cold Leg
3-RHS-MV8701C	RCS Loop 1, Hot Leg to RHR
3-RHS-MV8702C	RCS Loop 4, Hot Leg to RHR
3-RHS-MV8701A	RCS Loop 1, Hot Leg to RHR
3-RHS-MV8702B	RCS Loop 4, Hot Leg to RHR

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

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

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

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psia, 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 psia 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*	$<$ 0.10 ppm	$\leq$ 1.00 ppm
Chloride	$<$ 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 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

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

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

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, 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.

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

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.

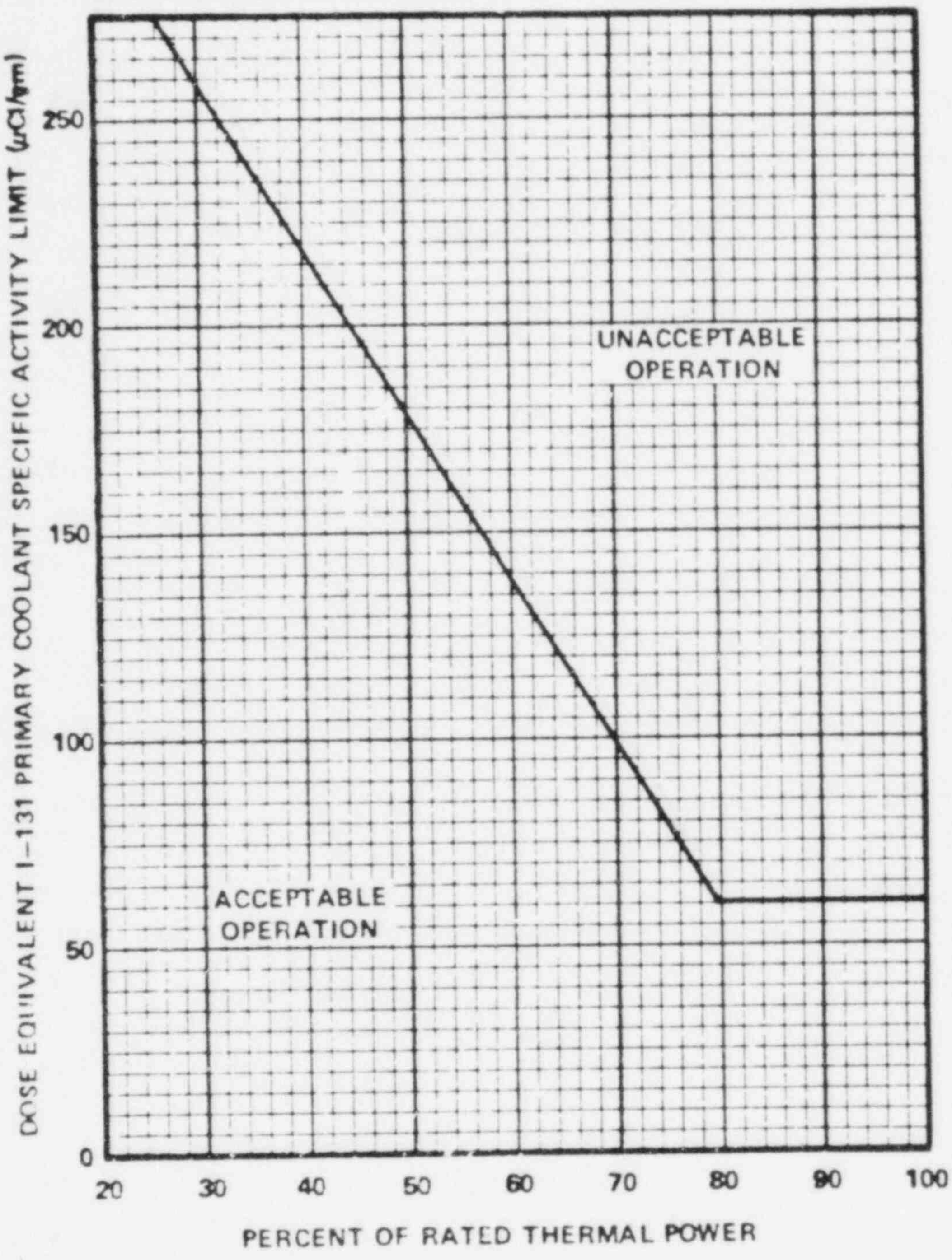


FIGURE 3.4-1

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 $\bar{E}$ Determination*	1 per 6 months**	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT I-131 or $100/\bar{E}$ 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

*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.

**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.

#Until the specific activity of the Reactor Coolant System is restored within its limits.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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 5°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILI At all times.

ACTION:

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

CONTROLLING MATERIAL	PLATE METAL
COPPER CONTENT	CONSERVATIVELY ASSUMED TO BE 0.10 WT%
PHOSPHORUS CONTENT	0.010 WT%
RT_{NDT} INITIAL	60°F
RT_{NDT} AFTER 10 EFPY	1/4T, 122°F 3/4T, 101°F

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

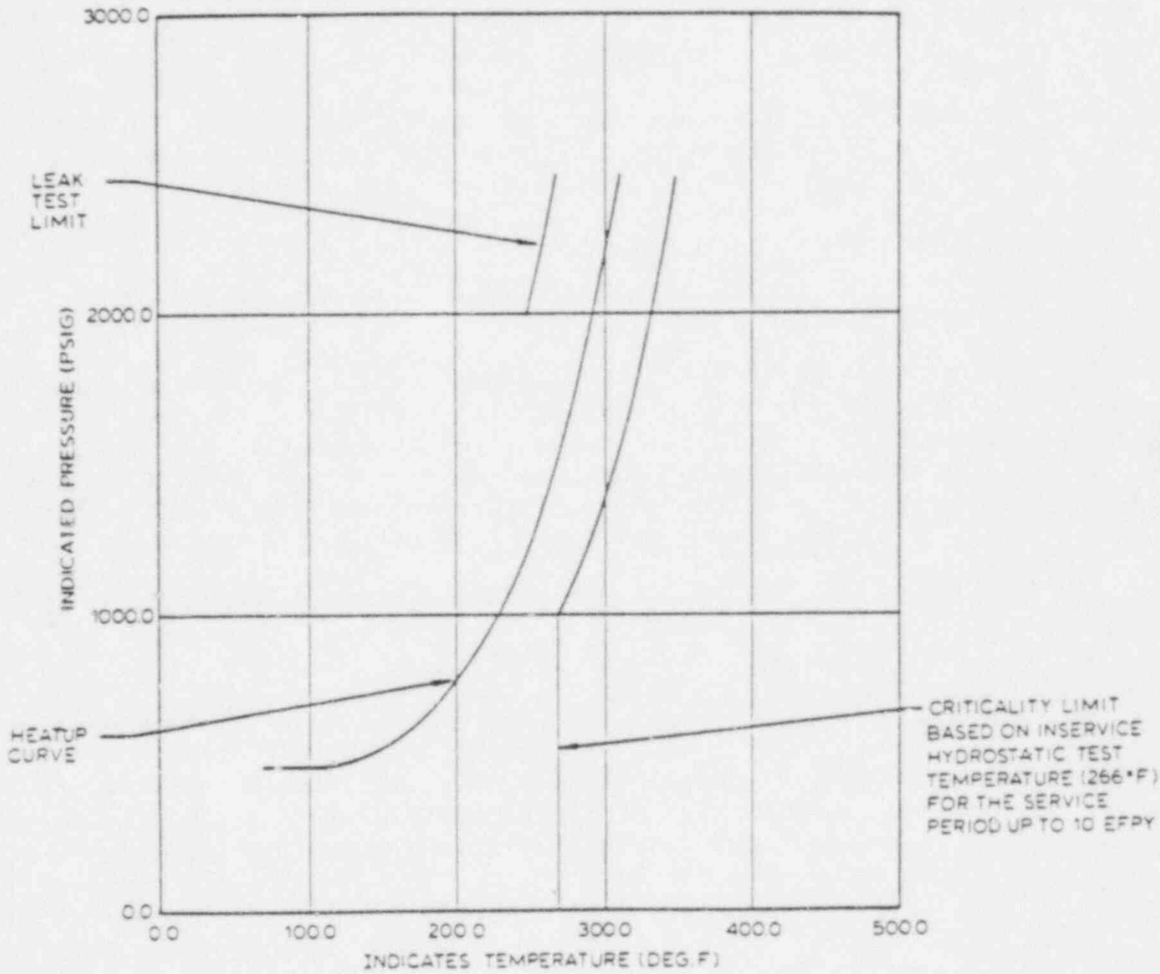


FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 10 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	PLATE METAL
COPPER CONTENT	CONSERVATIVELY ASSUMED TO BE 0.10 WT %.
PHOSPHORUS CONTENT	0.010 WT %.
RT_{NDT} INITIAL	60°F
RT_{NDT} AFTER 10 EFPY	1/4T, 122°F 3/4T, 101°F

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

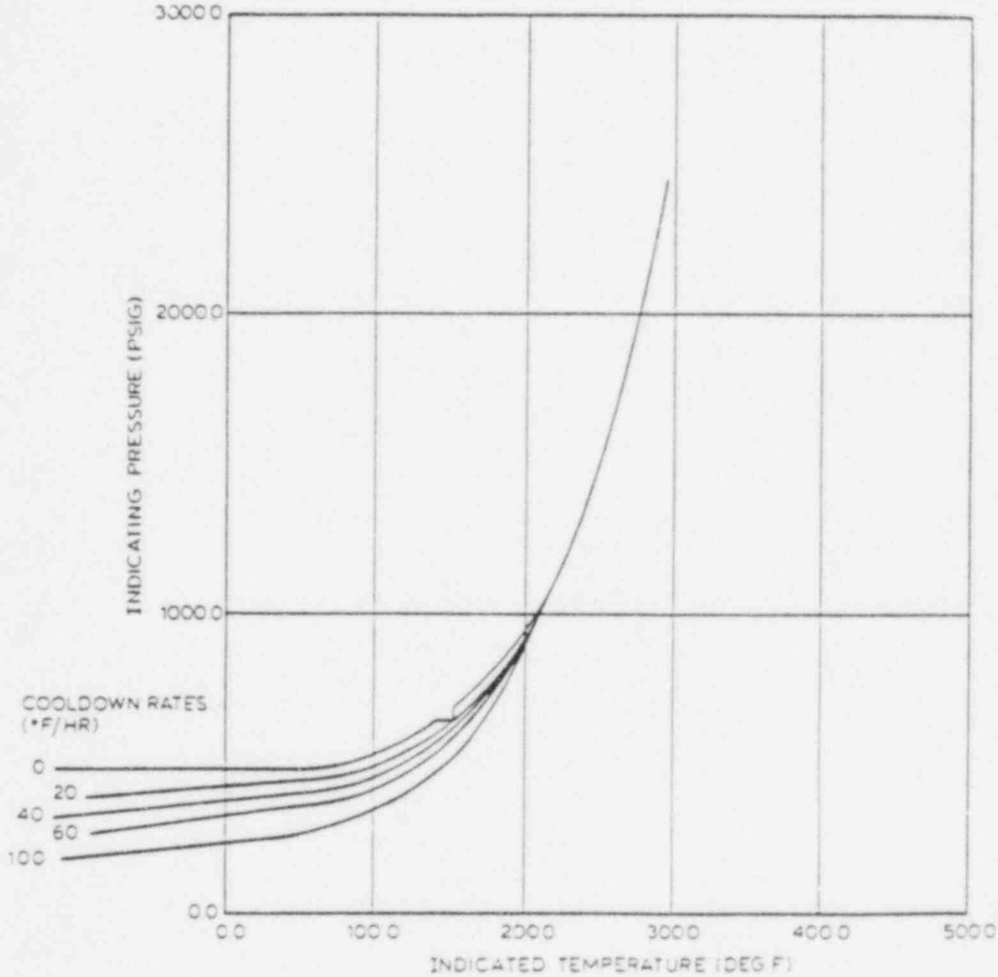


FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 10 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	58.5°	4.00	First Refueling
Y	241°	3.69	5
V	61°	3.69	9
X	238.5°	4.00	15
W	121.5°	4.00	STANDBY
Z	301.5°	4.00	STANDBY

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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 residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig, or

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

c. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 7.0 square inches.

APPLICABILITY: MODE 3 when the temperature of any RCS cold leg is less than or equal to 350°F, MODES 4 and 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one required PORV inoperable or one required RHR suction relief valve inoperable, restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 7.0 square inch vent within the next 8 hours.

b. With both required PORVs inoperable, within the next 8 hours either restore both RHR suction relief valves to OPERABLE status or depressurize and vent the RCS through at least a 7.0 square inch vent.

c. With both required RHR suction relief valves inoperable, within the next 8 hours either restore both PORVs to OPERABLE status or depressurize and vent the RCS through at least a 7.0 square inch vent.

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

e. 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 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 3RHS*RV8708A, by verifying at least once per 12 hours that 3RHS*MV8701A and 3RHS*MV8701C are open;

b. For RHR suction relief valve 3RHS*RV8708B, by verifying at least once per 12 hours that 3RHS*MV8702B and 3RHS*MV8702C are open; and

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS 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.

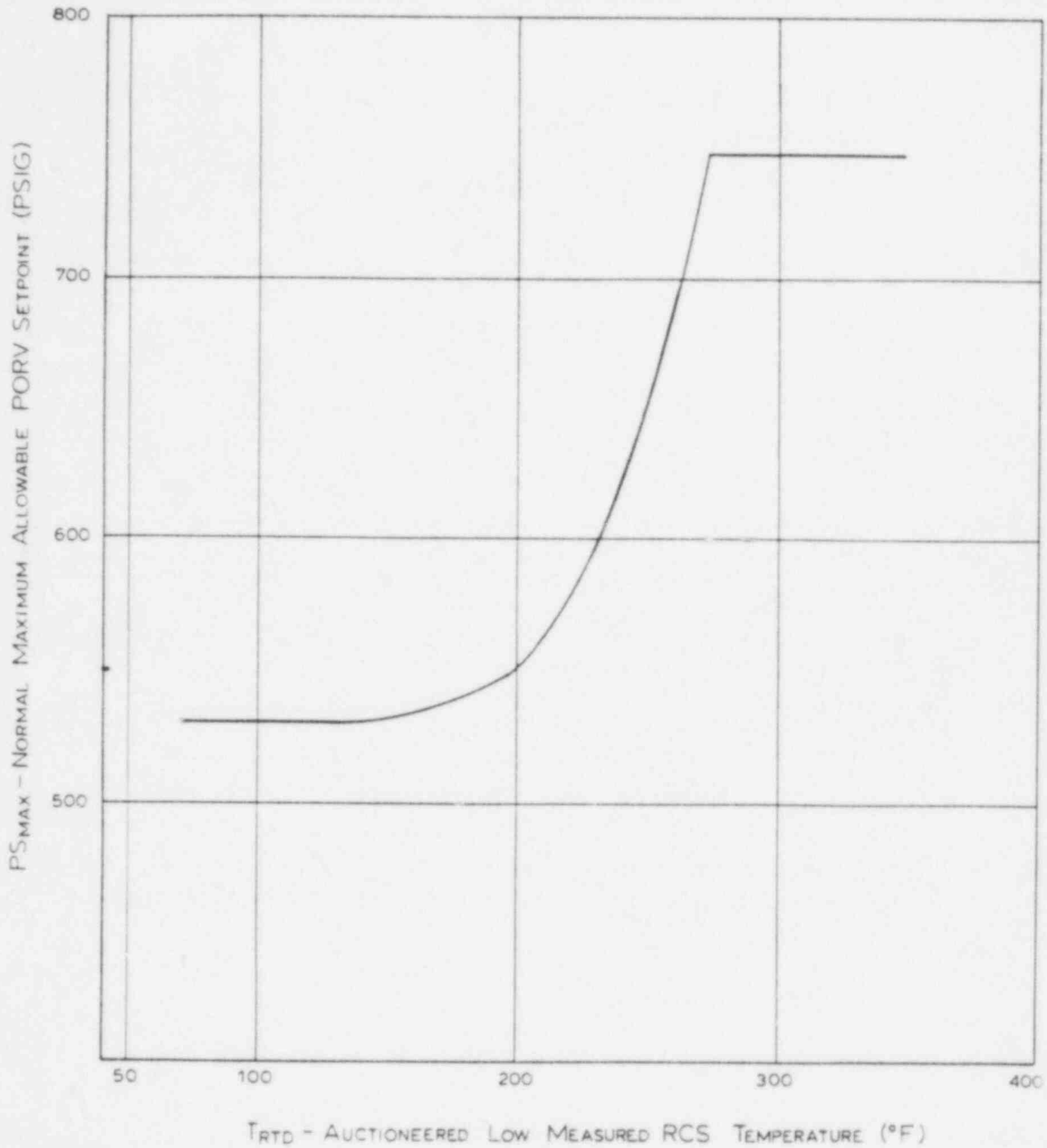


FIGURE 3.4-4a
NOMINAL MAXIMUM ALLOWABLE PORV
SETPOINT FOR THE COLD OVERPRESSURE SYSTEM
(FOUR LOOP OPERATION)

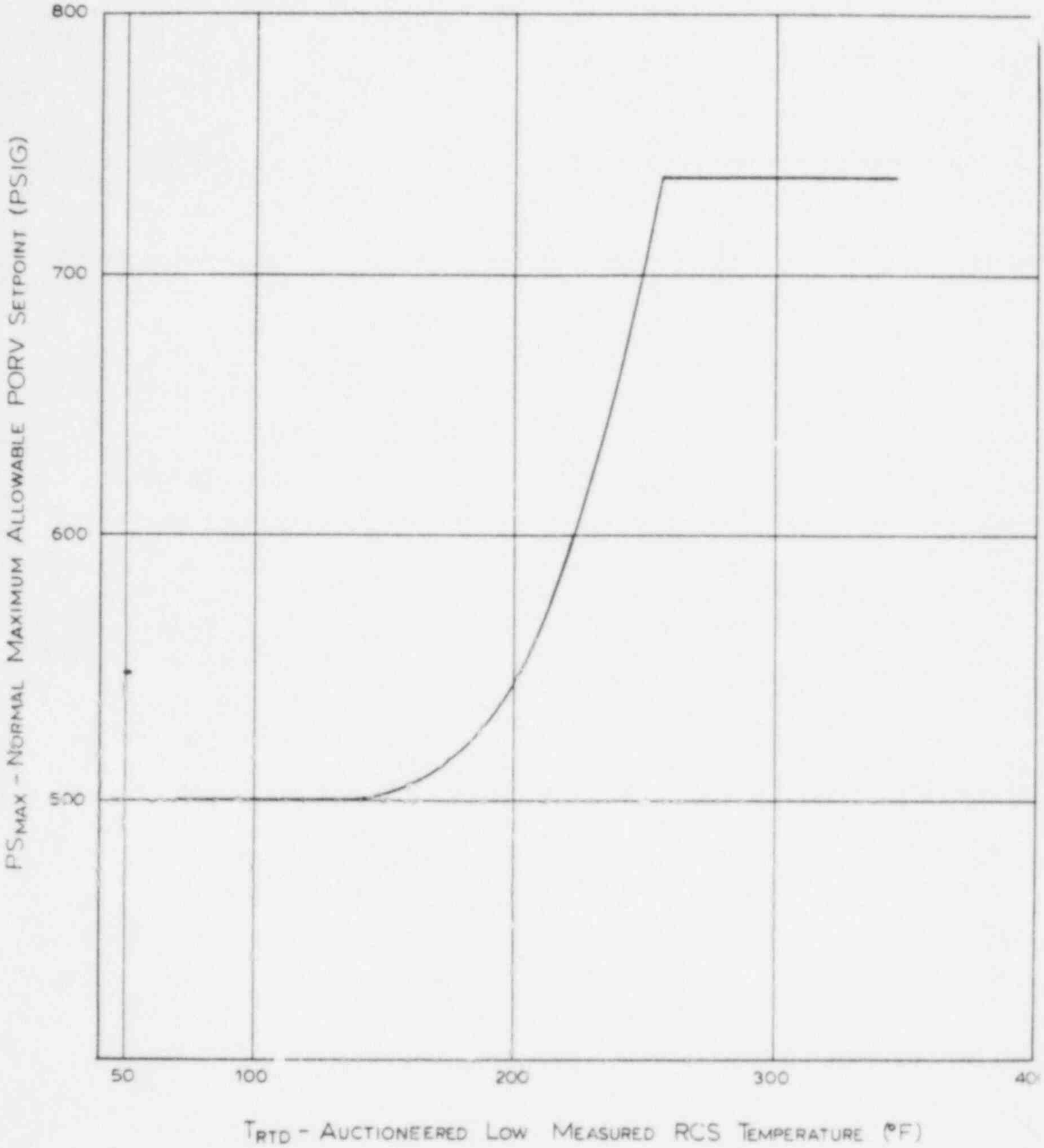


FIGURE 3.4-4b
NOMINAL MAXIMUM ALLOWABLE PORV
SETPOINT FOR THE COLD OVERPRESSURE SYSTEM
(THREE LOOP OPERATION)

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.

REACTOR COOLANT SYSTEM

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 two vent valves(s) and one block valve powered from emergency busses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and

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 vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each Reactor Coolant System vent path block valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel from the control room.

4.4.11.2 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,

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

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6618 and 6847 gallons,

c. A boron concentration of between 1900 and 2200 ppm, and

d. A nitrogen cover-pressure of between 636 and 694 psia.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one 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 reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks to be within the above limits, and

2) Verifying that each accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution; and

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit.

4.5.1.2 Each 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.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 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 RHR heat exchanger,

d. One OPERABLE RHR pump,

e. One OPERABLE containment recirculation heat exchanger,

f. One OPERABLE containment recirculation pump, and

g. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and capable of automatically stopping the RHR pump and being manually realigned to transfer 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
3SIH*MV8806	RWST Supply to SI Pumps	OPEN
3SIH*MV8802A	SI Pump A to Hot Leg Injection	CLOSED
3SIH*MV8802B	SI Pump B to Hot Leg Injection	CLOSED
3SIH*MV8835	SI Cold Leg Master Isolation	OPEN
3SIH*MV8813	SI Pump Master Miniflow Isolation	OPEN
3SIL*MV8840	RHR to Hot Leg Injection	CLOSED
3SIL*MV8809A	RHR Pump A to Cold Leg Injection	OPEN
3SIL*MV8809B	RHR Pump B to Cold Leg Injection	OPEN

b. At least once per 31 days by:

1) Verifying that the ECCS piping, except for the RSS pump, heat exchanger and associated 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 RHR 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 390 psia 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 765 psia the interlocks will cause the valves to automatically close.

SURVEILLANCE REQUIREMENTS (Continued)

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection actuation test signal, and

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

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

3) Verifying that the Residual Heat Removal pumps stop automatically upon receipt of a Low-Low RWST Level test signal.

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

1) Centrifugal charging pump $\geq$ 2411 psid,

2) Safety Injection pump $\geq$ 1470 psid,

3) RHR pump $\geq$ 165 psid, and

4) Containment recirculation pump $\geq$ 130 psid.

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

ECCS Throttle Valves Valve Number	Valve Number
3SIH*V6	3SIH*V25
3SIH*V7	3SIH*V27

SURVEILLANCE REQUIREMENTS (Continued)

ECCS Throttle Valves

Valve Number	Valve Number
3SIH*V8	3SIH*V107
3SIH*V9	3SIH*V108
3SIH*V21	3SIH*V109
3SIH*V23	3SIH*V111

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 339 gpm, and

b) The total pump flow rate is less than or equal to 560 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 463 gpm, and

b) The total pump flow rate is less than or equal to 670 gpm for the A pump and 650 gpm for the B pump.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3976 gpm.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 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 RHR heat exchanger,

c. One OPERABLE RHR pump,

d. One OPERABLE containment recirculation heat exchanger,

e. One OPERABLE containment recirculation pump, and

f. 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 the centrifugal charging pump, the containment recirculation pump, the containment recirculation heat exchanger, the flow path from the refueling water storage tank, or the flow path capable of taking suction from the containment sump, 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 RHR 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 RCS cold legs is less than or equal to 350°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 pumps, shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A contained borated water volume between 1,166,000 and 1,207,000 gallons,

b. A boron concentration between 2000 and 2200 ppm of boron,

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

d. A maximum solution temperature of 50°F.

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

ACTION

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

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, 54.1 psia (39.4 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 0.60 L_a.

---

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.9% by weight of the containment air per 24 hours at P_a, 54.1 psia (39.4 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 leakage rate of less than or equal to 0.01 L_a for all penetrations identified in Table 3.6-1 as Enclosure Building bypass leakage paths when pressurized to P_a.

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

ACTION:

With the measured overall integrated containment leakage rate exceeding 0.75 L_a, or the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, or the combined bypass leakage rate exceeding 0.01 L_a, restore the overall integrated leakage rate to less than 0.75 L_a, the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.01 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 methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a, 54.1 psia (39.4 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 0.75 L_a, 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 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

SURVEILLANCE REQUIREMENTS (Continued)

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 results, 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;

2) Has a duration sufficient to establish accurately the change in leakage rate betwern 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.

d. Type B and C tests shall be conducted with gas at P_a, 54.1 psia (34.9 psig), at intervals no greater than 24 months[a] except for tests involving:

1) Air locks, and

2) Purge supply and exhaust isolation valves with resilient material seals.

e. The combined bypass leakage rate shall be determined to be less than or equal to 0.01 L_a by applicable Type B and C tests at least once per 24 months except 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, 54.1 psig (39.4 psig), during each Type A test;

f. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

g. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2; and

h. The provisions of Specification 4.0.2 are not applicable.

TABLE 3.6-1

ENCLOSURE BUILDING BYPASS LEAKAGE PATHS

PENETRATION	RELEASE LOCATION
14	Ground Release
15	Ground Release
28	Plant Vent
29	Plant Vent
35	Plant Vent
36	Plant Vent
37	Plant Vent
38	Plant Vent
45	Plant Vent
52	Turbine Building Roof Exhaust
54	Turbine Building Roof Exhaust
56	Ground Release
70	Ground Release
72	Plant Vent
85	Ground Release
86	Plant Vent
116	Plant Vent

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 The 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, 54.1 psia (39.4 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.

---

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

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 no detectable seal leakage by pressure decay when the volume between the door seals is pressurized to greater than or equal to P_a, 54.1 psia (39.4 psig), for at least 15 minutes;

b. By conducting overall air lock leakage tests at not less than P_a, 54.1 psia (39.4 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.

*The provisions of Specification 4.0.2 are not applicable.
**This represents an exemption to Appendix J, paragraph III.D.2.(b)(ii), of 10 CFR Part 50.

CONTAINMENT SYSTEMS

AIR PARTIAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment air partial pressure shall be maintained greater than or equal to 8.9 psia and within the acceptable operation range shown on Figure 3.6-1 as a function of service water temperature.

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

ACTION:

With the containment air partial pressure less than 8.9 psia or above the service water temperature limit line shown on Figure 3.6-1, restore the air partial 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 air partial pressure shall be determined to be within the limits at least once per 12 hours.

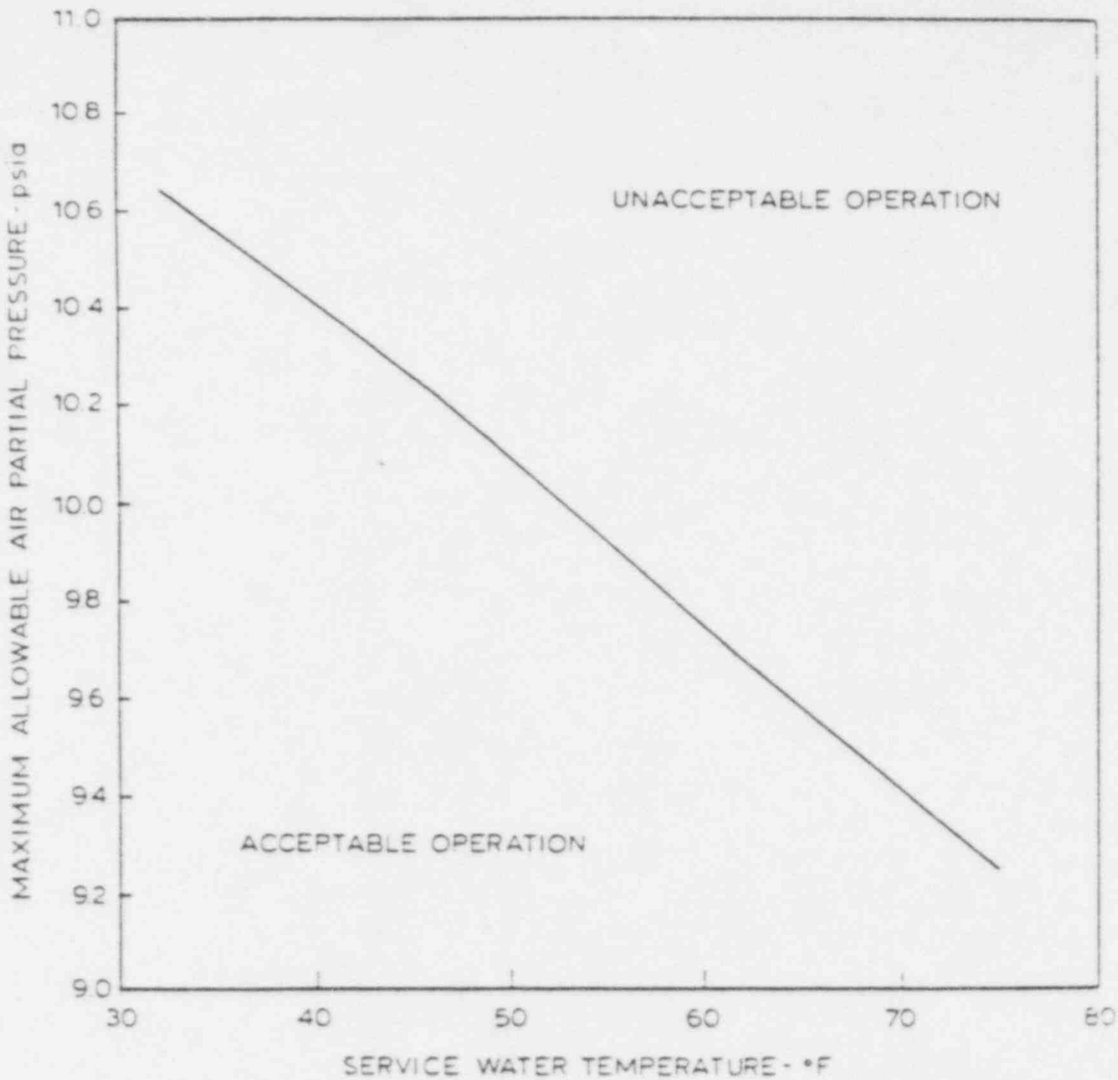


FIGURE 3.6-1

MAXIMUM ALLOWABLE PRIMARY CONTAINMENT AIR PARTIAL PRESSURE VERSUS SERVICE WATER TEMPERATURE

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained greater than or equal to 80°F and less than or equal to 120°F.

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

ACTION:

With the containment average air temperature less than 80°F or greater than 120°F, restore the average air temperature to within the limit within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. 94 ft elevation, E outside crane wall

b. 86 ft elevation, NW outside crane wall

c. 75 ft elevation, W Steam Generator platform

d. 75 ft elevation, E Steam Generator platform

e. 45 ft elevation, Pressurizer cubicle, crane wall

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment 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 not conforming to the above requirements, restore the structural integrity to within the limits 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

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, 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 these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and each 42-inch containment shutdown purge supply and exhaust isolation valve shall be closed and locked closed.

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

ACTION:

a. With a 42-inch containment purge supply and/or exhaust isolation valve open or not locked closed, close and/or lock close that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The containment purge supply and exhaust isolation valves shall be verified to be locked closed and closed at least once per 31 days.

4.6.1.7.2 At least once per 6 months on a STAGGERED TEST BASIS, the inboard and outboard valves with resilient material seals in each locked closed 42-inch containment purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.005 L_a when pressurized to P_t, 34.4 psia (19.7 psig).

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Quench Spray subsystems shall be OPERABLE.

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

ACTION:

With one Containment Quench Spray subsystem 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

4.6.2.1 Each Containment Quench Spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days:

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

2) Verifying the temperature of the borated water in the refueling water storage tank is between 40°F and 50°F

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 114 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 CDA test signal, and

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

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.

RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent Recirculation Spray Systems shall be OPERABLE.

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

ACTION:

With one Recirculation Spray System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Recirculation Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each Recirculation 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 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 130 psid when tested pursuant to Specification 4.0.5;

c. At least once per 18 months by verifying that on a CDA test signal, each recirculation spray pump starts automatically after a 660 ± 20 second delay;

d. 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 CDA test signal; and

e. 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.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The Spray Additive System shall be OPERABLE with:

a. A chemical addition tank containing a volume of between 19,100 and 20,100 gallons of between 1.35 and 2.00% by weight NaOH solution, and

b. Two gravity feed paths each capable of adding NaOH solution from the chemical addition tank to each Containment Quench Spray subsystem pump suction.

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The Spray Additive System shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis is within the above limits.

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 CDA test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-2 shall be OPERABLE with isolation times as shown in Table 3.6-2.

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

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-2 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 Table 3.6-2 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-2 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, and

c. Verifying that on a Containment High Radiation test signal, each purge supply and exhaust isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve of Table 3.6-2 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase A Isolation		
3SSR-CTV26	Reactor Coolant Hot Leg Sample (Inside)	<60
3SSR-CTV27	Reactor Coolant Hot Leg Sample (Outside)	<60
3SSR-CTV22	PZR Liquid Sample (Inside)	<60
3SSR-CTV23	PZR Liquid Sample (Outside)	<60
3SSR-CTV20	PZR Vapor Space Sample (Inside)	<60
3SSR-CTV21	PZR Vapor Space Sample (Outside)	<60
3SSR-CV8026	PRT Gas Sample (Inside)	<60
3SSR-CV8025	PRT Gas Sample (Outside)	<60
3SSR-CTV29	Reactor Coolant Cold Leg Sample (Inside)	<60
3SSR-CTV30	Reactor Coolant Cold Leg Sample (Outside)	<60
3SSR-CTV32	S.I. Accumulator Sample (Inside)	<60
3SSR-CTV33	S.I. Accumulator Sample (Outside)	<60
3SIL-CV8968	Nitrogen to S.I. Accumulators (Inside)	<60
3SIL-CV8880	Nitrogen to S.I. Accumulators (Outside)	<60
3PGS-CV8046	Primary Grade Water to PRT (Inside)	<60

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase A Isolation (Continued)		
3PGS-CV8028	Primary Grade Water to PRT (Outside)	<60
3CHS-MV8112	Seal Water Return From RCPs (Inside)	<60
3CHS-MV8100	Seal Water Return From RCPs (Outside)	<60
3CHS-CV8160	Reactor Coolant Letdown (Inside)	<60
3CHS-CV8152	Reactor Coolant Letdown (Outside)	<60
3DGS-CTV24	PRT and CTMT Drains Transfer Pumps Discharge (Inside)	<60
3DGS-CTV25	PRT and CTMT Drains Transfer Pumps Discharge (Outside)	<60
3DAS-CTV24	CTMT Drains Sump Pump Discharge (Inside)	<60
3DAS-CTV25	CTMT Drains Sump Pump Discharge (Outside)	<60
3VRS-CTV20	PRT and CTMT Drains Transfer Tank Vent (Inside)	<60
3VRS-CTV21	PRT and CTMT Drains Transfer Tank Vent (Outside)	<60
3CVS-CTV20A	CTMT Vacuum Pump Suction (Outside)	<60
3CVS-CTV20B	CTMT Vacuum Pump Suction (Outside)	<60
3CVS-CTV21A	CTMT Vacuum Pump Suction (Outside)	<60

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase A Isolation (Continued)		
3CVS-CTV21B	CTMT Vacuum Pump Suction (Outside)	<60
3CDS-CTV91A	Chill Water Supply (Inside)	<60
3CDS-CTV91B	Chill Water Supply (Inside)	<60
3CDS-CTV38A	Chill Water Supply (Outside)	<60
3CDS-CTV38B	Chill Water Supply (Outside)	<60
3CDS-CTV40A	Chill Water Return (Inside)	<60
3CDS-CTV40B	Chill Water Return (Inside)	<60
3CDS-CTV39A	Chill Water Return (Outside)	<60
3CDS-CTV39B	Chill Water Return (Outside)	<60
3IAS-MOV72	Instrument Air (Inside)	<60
3IAS-PV15	Instrument Air (Outside)	<60
3FPW-CTV49	Fire Protection (Inside)	<60
3FPW-CTV48	Fire Protection (Outside)	<60
3CMS-MOV24	CTMT Atmosphere Monitor Discharge (Inside)	<60
3CMS-CTV23	CTMT Atmosphere Monitor Discharge (Outside)	<60

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase A Isolation (Continued)		
3CMS-CTV20	CTMT Atmosphere Monitor Suction (Outside)	<60
3CMS-CTV21	CTMT Atmosphere Monitor Suction (Outside)	<60
3SIH-CV8871	S.I. Test and Accumulator Fill (Inside)	<60
3SIH-CV8964	S.I. Test and Accumulator Fill (Outside)	<60
3SIH-CV8888	S.I. Test and Accumulator Fill (Outside)	<60
3GSN-CTV105	Nitrogen Supply Header (Inside)	<60
3GSN-CV8033	Nitrogen Supply Header (Outside)	<60
3SSP-CTV7	Post-Accident Sample (Inside)	<60
3SSP-CTV8	Post-Accident Sample Return (Inside)	<60
3SIH-CV8843	High-Pressure Boron Injection (Inside)	<60
3SIL-CV8890A	RHR Cold Leg Injection to Test (Inside)	<60
3SIL-CV8890B	RHR Cold Leg Injection to Test (Inside)	<60

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase A Isolation (Continued)		
3SIL-CV8825	RHR Hot Leg Injection (Inside)	<60
3SIH-CV8881	S.I. Pump Hot Leg Injection (Inside)	<60
3SIH-CV8824	S.I. Pump Hot Leg Injection (Inside)	<60
3SIH-CV8823	S.I. Pump Cold Leg Injection	<60
2. Phase B Isolation		
3CCP-MOV45A	RPCCW CTMT Supply (Outside)	<60
3CCP-MOV45B	RPCCW CTMT Supply (Outside)	<60
3CCP-MOV48A	RPCCW CTMT Return (Inside)	<60
3CCP-MOV48B	RPCCW CTMT Return (Inside)	<60
3CCP-MOV49A	RPCCW CTMT Return (Outside)	<60
3CCP-MOV49B	RPCCW CTMT Return (Outside)	<60

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves		
3CHS-MV8105	Reactor Coolant Charging	<10
3CVS-MOV25	CTMT Vacuum Pump Discharge (Inside)	NSR
3CHS-MV8109A	RCP Seal Injection Isolation (Outside)	NSR
3CHS-MV8109B	RCP Seal Injection Isolation (Outside)	NSR
3CHS-MV8109C	RCP Seal Injection Isolation (Outside)	NSR
3CHS-MV8109D	RCP Seal Injection Isolation (Outside)	NSR
3FWA-MOV35A	Motor-Driven Aux. Feedwater Header Isolation (Outside)	NSR
3FWA-MOV35B	Motor-Driven Aux. Feedwater Header Isolation (Outside)	NSR
3FWA-MOV35C	Motor-Driven Aux. Feedwater Header Isolation (Outside)	NSR
3FWA-MOV35D	Motor-Driven Aux. Feedwater Header Isolation (Outside)	NSR
3LMS-MOV40A	CTMT Pressure Instrument Isolation (Outside)	NSR
3LMS-MOV40B	CTMT Pressure Instrument Isolation (Outside)	NSR
3LMS-MOV40C	CTMT Pressure Instrument Isolation (Outside)	NSR

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves (Continued)		
3LMS-MOV40D	CTMT Pressure Instrument Isolation (Outside)	NSR
3MSS-MOV74A	Main Steam Pressure Relieving Bypass (Outside)	NSR
3MSS-MOV74B	Main Steam Pressure Relieving Bypass (Outside)	NSR
3MSS-MOV74C	Main Steam Pressure Relieving Bypass (Outside)	NSR
3MSS-MOV74D	Main Steam Pressure Relieving Bypass (Outside)	NSR
3RHS-MV8701A	Hot Leg to RHS Suction Isolation (Inside)	NSR
3RHS-MV8701B	Hot Leg to RHS Suction Isolation (Outside)	NSR
3RHS-MV8702A	Hot Leg to RHS Suction Isolation (Outside)	NSR
3RHS-MV8702B	Hot Leg to RHS Suction Isolation (Inside)	NSR

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves (Continued)		
3RSS-MOV20A	CTMT Recirculation Pump Discharge (Outside)	NSR
3RSS-MOV20B	CTMT Recirculation Pump Discharge (Outside)	NSR
3RSS-MOV20C	CTMT Recirculation Pump Discharge (Outside)	NSR
3RSS-MOV20D	CTMT Recirculation Pump Discharge (Outside)	NSR
3RSS-MOV23A	CTMT Recirculation Pump Suction (Outside)	NSR
3RSS-MOV23B	CTMT Recirculation Pump Suction (Outside)	NSR
3RSS-MOV23C	CTMT Recirculation Pump Suction (Outside)	NSR
3RSS-MOV23D	CTMT Recirculation Pump Suction (Outside)	NSR
3QSS-MOV34A	Quench Spray Pump Discharge (Outside)	NSR
3QSS-MOV34B	Quench Spray Pump Discharge (Outside)	NSR
3SIH-MV8801A	Charging S.I. Header Isolation (Outside)	NSR
3SIH-MV8801B	Charging S.I. Header Isolation (Outside)	NSR
3SIL-MV8809A	RHS Cold Leg Injection Header Isolation (Outside)	NSR

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves (Continued)		
3SIL-MV8809B	RHS Cold Leg Injection Header Isolation (Outside)	NSR
3SIL-MV8840	RHS Hot Leg Injection Header Isolation (Outside)	NSR
3SIH-MV8802A	S.I. Hot Leg Injection Header Isolation (Outside)	NSR
3SIH-MV8802B	S.I. Hot Leg Injection Header Isolation (Outside)	NSR
3SIH-MV8835	S.I. Cold Leg Injection Header Isolation (Outside)	NSR
3MSS-CTV27A	Main Steam Isolation Trip Valves (Outside)	<5
3MSS-CTV27B	Main Steam Isolation Trip Valves (Outside)	<5
3MSS-CTV27C	Main Steam Isolation Trip Valves (Outside)	<5
3MSS-CTV27D	Main Steam Isolation Trip Valves (Outside)	<5

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves (Continued)		
3MSS-HV28A	Main Steam Isolation Bypass (Outside)	<10
3MSS-HV28B	Main Steam Isolation Bypass (Outside)	<10
3MSS-HV28C	Main Steam Isolation Bypass (Outside)	<10
3MSS-HV28D	Main Steam Isolation Bypass (Outside)	<10
3DTM-AOV63A	AFW Turbine Steam Line Drains (Outside)	<10
3DTM-AOV63B	AFW Turbine Steam Line Drains (Outside)	<10
3DTM-AOV63D	AFW Turbine Steam Line Drains (Outside)	<10
3DTM-AOV64A	AFW Turbine Steam Line Drains (Outside)	<10
3DTM-AOV64B	AFW Turbine Steam Line Drains (Outside)	<10
3DTM-AOV64D	AFW Turbine Steam Line Drains (Outside)	<10
3DTM-AOV29A	Main Steam Line Drains Upstream MSITV (Outside)	<10
3DTM-AOV29B	Main Steam Line Drains Upstream MSITV (Outside)	<10
3DTM-AOV29C	Main Steam Line Drains Upstream MSITV (Outside)	<10
3DTM-AOV29D	Main Steam Line Drains Upstream MSITV (Outside)	<10

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves (Continued)		
3DTM-AOV61A	Main Steam Line Drains Upstream MSITV (Outside)	<10
3DTM-AOV61B	Main Steam Line Drains Upstream MSITV (Outside)	<10
3DTM-AOV61C	Main Steam Line Drains Upstream MSITV (Outside)	<10
3DTM-AOV61D	Main Steam Line Drains Upstream MSITV (Outside)	<10
3MSS-PV20A	Main Steam Pressure Relieving Control Valve (Outside)	NSR
3MSS-PV20B	Main Steam Pressure Relieving Control Valve (Outside)	NSR
3MSS-PV20C	Main Steam Pressure Relieving Control Valve (Outside)	NSR
3MSS-PV20D	Main Steam Pressure Relieving Control Valve (Outside)	NSR
3MSS-AOV31A	AFW Turbine Steam Line Isolation (Outside)	NSR
3MSS-AOV31B	AFW Turbine Steam Line Isolation (Outside)	NSR

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves (Continued)		
3MSS-AOV31D	AFW Turbine Steam Line Isolation (Outside)	NSR
3FWS-CTV41A	Feedwater Isolation Trip Valve (Outside)	<5
3FWS-CTV41B	Feedwater Isolation Trip Valve (Outside)	<5
3FWS-CTV41C	Feedwater Isolation Trip Valve (Outside)	<5
3FWS-CTV41D	Feedwater Isolation Trip Valve (Outside)	<5
3BDG-CTV22A	Blowdown Isolation Trip Valve (Outside)	<10
3BDG-CTV22B	Blowdown Isolation Trip Valve (Outside)	<10
3BDG-CTV22C	Blowdown Isolation Trip Valve (Outside)	<10
3BDG-CTV22D	Blowdown Isolation Trip Valve (Outside)	<10
3SSR-CTV19A	Blowdown Sample Isolation Trip Valve (Outside)	<10
3SSR-CTV19B	Blowdown Sample Isolation Trip Valve (Outside)	<10
3SSR-CTV19C	Blowdown Sample Isolation Trip Valve (Outside)	<10
3SSR-CTV19D	Blowdown Sample Isolation Trip Valve (Outside)	<10

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves (Continued)		
3FWA-HV36A	Turbine-Driven Aux Feedwater Header Isolation (Outside)	NSR
3FWA-HV36B	Turbine-Driven Aux Feedwater Header Isolation (Outside)	NSR
3FWA-HV36C	Turbine-Driven Aux Feedwater Header Isolation (Outside)	NSR
3FWA-HV36D	Turbine-Driven Aux Feedwater Header Isolation (Outside)	NSR
3CVS-AOV23	CTMT Vacuum Ejector Suction Isolation (Inside)	NSR
3HVU-CTV33A	CTMT Purge Isolation Trip Valve (Inside)	<3
3HVU-CTV32A	CTMT Purge Isolation Trip Valve (Outside)	<3
3HVU-CTV33B	CTMT Purge Isolation Trip Valve (Inside)	<3
3HVU-CTV32B	CTMT Purge Isolation Trip Valve (Outside)	<3
3SGF-AOV24A	Steam Generator Chem. Feed Isolation (Outside)	<60

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
3. Remote-Operated Valves (Continued)		
3SGF-AOV24B	Steam Generator Chem. Feed Isolation (Outside)	<60
3SGF-AOV24C	Steam Generator Chem. Feed Isolation (Outside)	<60
3SGF-AOV24D	Steam Generator Chem. Feed Isolation (Outside)	<60
4. Manual Valves		
3FPW-V661*	Fire Protection (Inside)	NA
3FPW-V666*	Fire Protection (Outside)	NA
3SSP-V13*	Post-Accident Sample (Outside)	NA
3SSP-V14*	Post-Accident Sample Return (Outside)	NA
3HCS-V2*	DBA Hydrogen Recombiner Suction (Outside)	NA
3HCS-V3*	DBA Hydrogen Recombiner Suction (Outside)	NA
3HCS-V9*	DBA Hydrogen Recombiner Suction (Outside)	NA
3HCS-V10*	DBA Hydrogen Recombiner Suction (Outside)	NA

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
4. Manual Valves (Continued)		
3HCS-V6*	DBA Hydrogen Recombiner Discharge (Outside)	NA
3HCS-V13*	DBA Hydrogen Recombiner Discharge (Outside)	NA
3SAS-V875*	Service Air Line (Inside)	NA
3SAS-V50*	Service Air Line (Outside)	NA
3SFC-V991	Refueling Cavity Purification Inlet (Inside)	NA
3SFC-V992	Refueling Cavity Purification Inlet (Outside)	NA
3SFC-V990	Refueling Cavity Purification Outlet (Inside)	NA
3SFC-V989	Refueling Cavity Purification Outlet (Outside)	NA
3CHS-V371*	Reactor Coolant Loop Fill (Outside)	NA
3HVU-V5	Containment Purge Air (Outside)	NA
3CCP-V886*	Demineralized Water CTMT Supply (Inside)	NA
3CCP-V887*	Demineralized Water CTMT Supply (Outside)	NA
3CVS-V13*	Containment Vacuum Pump Discharge (Outside)	NA
3CVS-V20	Containment Vacuum Ejector Suction (Outside)	NA

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
5. Check Valves		
3CHS-V58	Charging to RCS (Inside)	NA
3CHS-V372	Reactor Coolant Loop Fill (Inside)	NA
3CHS-V394	Reactor Coolant Pump Seal Water Supply (Inside)	NA
3CHS-V434	Reactor Coolant Pump Seal Water Supply (Inside)	NA
3CHS-V467	Reactor Coolant Pump Seal Water Supply (Inside)	NA
3CHS-V501	Reactor Coolant Pump Seal Water Supply (Inside)	NA
3CCP-V18	Component Cooling Water CTMT Supply (Inside)	NA
3CCP-V60	Component Cooling Water CTMT Supply (Inside)	NA
3HCS-V7	DBA Hydrogen Recombiner Discharge (Inside)	NA
3HCS-V14	DBA Hydrogen Recombiner Discharge (Inside)	NA
3QSS-V4	Quench Spray Pump Discharge (Inside)	NA

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
5. Check Valves (Continued)		
3QSS-V8	Quench Spray Pump Discharge (Inside)	NA
3RSS-V3	CTMT Recirculation Pump Discharge (Inside)	NA
3RSS-V6	CTMT Recirculation Pump Discharge (Inside)	NA
3RSS-V9	CTMT Recirculation Pump Discharge (Inside)	NA
3RSS-V12	CTMT Recirculation Pump Discharge (Inside)	NA
3SGF-V29	Steam Generator Chemical Feed Supply (Inside)	NA
3SGF-V31	Steam Generator Chemical Feed Supply (Inside)	NA
3SGF-V33	Steam Generator Chemical Feed Supply (Inside)	NA
3SGF-V35	Steam Generator Chemical Feed Supply (Inside)	NA
3SIH-V5	High Pressure Boron Injection to Cold Legs (Inside)	NA
3SIH-V22	Safety Injection Cold Leg Discharge (Inside)	NA

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
5. Check Valves (Continued)		
3SIH-V24	Safety Injection Cold Leg Discharge (Inside)	NA
3SIH-V26	Safety Injection Cold Leg Discharge (Inside)	NA
3SIH-V28	Safety Injection Cold Leg Discharge (Inside)	NA
3SIH-V110	Safety Injection Hot Leg Discharge (Inside)	NA
3SIH-V112	Safety Injection Hot Leg Discharge (Inside)	NA
3SIL-V6	RHR Discharge to Cold Legs (Inside)	NA
3SIL-V7	RHR Discharge to Cold Legs (Inside)	NA
3SIL-V12	RHR Discharge to Cold Legs (Inside)	NA
3SIL-V13	RHR Discharge to Cold Legs (Inside)	NA
3SIL-V26	RHR Discharge to Hot Legs (Inside)	NA
3SIL-V28	RHR Discharge to Hot Legs (Inside)	NA
3SIL-V27	Safety Injection Hot Leg Discharge (Inside)	NA
3SIL-V29	Safety Injection Hot Leg Discharge (Inside)	NA

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
6. Relief Valves		
3CHS-RV8113	RCP Seal Water Return (Inside)	NA
3CHS-RV8117	Reactor Coolant Letdown (Inside)	NA
3MSS-RV22A, B, C, D	Main Steam Line Safety Valves (Outside)	NA
3MSS-RV23A, B, C, D	Main Steam Line Safety Valves (Outside)	NA
3MSS-RV24A, B, C, D	Main Steam Line Safety Valves (Outside)	NA
3MSS-RV25A, B, C, D	Main Steam Line Safety Valves (Outside)	NA
3MSS-RV26A, B, C, D	Main Steam Line Safety Valves (Outside)	NA
3RHS-RV8708A	RHR Pump Hot Leg Suction (Inside)	NA
3RHS-RV8708B	RHR Pump Hot Leg Suction (Inside)	NA

*May be opened on an intermittent basis under administrative control.

NOTES:

1. The maximum closure times shown are based on limits set by off-site dose calculations.
2. NSR - No Stroke Time Required. This applies to valves which open, go open or are opened during an accident, or valves which are normally shut and stay shut during an accident.
3. NA - Closure time not applicable. This applies to manual valves, check valves and relief valves.

CONTAINMENT SYSTEMS

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 sample gas containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

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 Hydrogen Recombiner System functional test that the minimum reaction chamber gas temperature increases to greater than or equal to 700°F within 90 minutes and is maintained for at least 2 hours and that the purge blower operates for 15 minutes.

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 recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.),

   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 10,000 ohms, and

   4) Verifying during a recombiner system functional test using containment atmospheric air at a flow rate of greater than or equal to 50 scfm, that the gas temperature increases to greater than or equal to 1100°F within 5 hours and is maintained for at least 4 hours.

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

LIMITING CONDITION FOR OPERATION

3.6.5.1 The inside and outside isolation valves in the steam jet air ejector suction line shall be closed.

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

ACTION:

With the inside or outside isolation valves in the steam jet air ejector suction line not closed, restore the valve to the closed position 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

4.6.5.1.1 The steam jet air ejector suction line outside isolation valve shall be determined to be in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 200°F and at least once per 31 days thereafter.

4.6.5.1.2 The steam jet air ejector suction line inside isolation valve shall be determined to be locked in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 200°F.

3/4.6.6 SECONDARY CONTAINMENT

SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Supplementary Leak Collection and Release Systems shall be OPERABLE.

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

ACTION:

With one Supplementary Leak Collection and Release 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 Supplementary Leak Collection and Release 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 charcoal adsorbers and verifying the system flowrate against the system pressure drop using certified fan curves and 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 system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% 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 9,500 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 0.175%; and

   3) Verifying a system flow rate of 9,500 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

SURVEILLANCE REQUIREMENTS (Continued)

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 0.175%:

d. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.25 inches Water Gauge while operating the system at a flow rate of 9,500 cfm ± 10% and verifying the fan curve for observed pressure drop against the system flowrate,

   2) Verifying that the system starts on a Safety Injection test signal,

   3) Verifying that each system produces a negative pressure of greater than or equal to 0.25 inch Water Gauge in the annulus within 50 seconds after a start signal, and

   4) Verifying that the heaters dissipate 50 ± 5 kW when tested in accordance with ANSI N510-1980.

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 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 9,500 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 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9,500 cfm ± 10%.

*ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

ENCLOSURE BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.2 ENCLOSURE BUILDING INTEGRITY shall be maintained.

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

ACTION:

Without ENCLOSURE BUILDING INTEGRITY, restore ENCLOSURE BUILDING INTEGRITY 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

4.6.6.2 ENCLOSURE BUILDING INTEGRITY shall be demonstrated at least once per 31 days by verifying that each door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

ENCLOSURE BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the structural integrity of the enclosure building not conforming to the above requirements, restore the structural integrity to within the limits 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

4.6.6.3 The structural integrity of the enclosure 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 enclosure building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the enclosure building detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days.

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 of an unisolated reactor coolant loop shall be OPERABLE with lift settings as specified in Table 3.7-3.

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. With three reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves associated with an operating loop 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 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING THREE 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	64
2	48
3	32

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (± 1%)*	ORIFICE SIZE
LOOP 1		
RV22A	1185 psig	16.0 square inches
RV23A	1195 psig	16.0 square inches
RV24A	1205 psig	16.0 square inches
RV25A	1215 psig	16.0 square inches
RV26A	1225 psig	16.0 square inches
LOOP 2		
RV22B	1185 psig	16.0 square inches
RV23B	1195 psig	16.0 square inches
RV24B	1205 psig	16.0 square inches
RV25B	1215 psig	16.0 square inches
RV26B	1225 psig	16.0 square inches
LOOP 3		
RV22C	1185 psig	16.0 square inches
RV23C	1195 psig	16.0 square inches
RV24C	1205 psig	16.0 square inches
RV25C	1215 psig	16.0 square inches
RV26C	1225 psig	16.0 square inches
LOOP 4		
RV22D	1185 psig	16.0 square inches
RV23D	1195 psig	16.0 square inches
RV24D	1205 psig	16.0 square inches
RV25D	1215 psig	16.0 square inches
RV26D	1225 psig	16.0 square inches

*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 within 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 on recirculation flow each motor-driven pump develops a differential pressure of greater than or equal to 1460 psid when tested pursuant to Specification 4.0.5;

2) Verifying that on recirculation flow the steam turbine-driven pump develops a differential pressure of greater than or equal to 1640 psid when the secondary steam supply pressure is greater than 800 psig. 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; and

4) Verifying that each auxiliary feedwater control and isolation valve in the flow path is in the fully open position when above 10% RATED THERMAL POWER.

b. At least once per 18 months during shutdown by verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation 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 flow to each steam generator.

PLANT SYSTEMS

DEMINERALIZED WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The demineralized water storage tank (DWST) shall be OPERABLE with a contained water volume of at least 334,000 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the DWST inoperable, within 4 hours either:

a. Restore the DWST 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 condensate storage tank (CST) as a backup supply to the auxiliary feedwater pumps and restore the DWST to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The DWST shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying that the combined volume of both the DWST and CST is at least 334,000 gallons of water whenever the condensate storage tank and DWST are the supply source for the auxiliary feedwater pumps.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microCurie/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	At 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.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV 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 MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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 REACTOR PLANT COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent reactor plant component cooling water safety loops shall be OPERABLE.

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

ACTION:

With only one reactor plant component cooling water safety 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 reactor plant component cooling water safety 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 actuates to its correct position on its associated Engineered Safety Feature actuation signal, and

   2) Each Component Cooling Water System pump starts automatically on an SIS test signal.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; 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 its associated Engineered Safety Feature actuation signal, and

   2) Each Service Water System pump starts automatically on an SIS test signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with an average water temperature of less than or equal to 75°F at the Unit 3 intake structure.

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

ACTION:

With the requirement 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 ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the average water temperature at the Unit 3 intake structure to be within limits.

b. At least once per 6 hours by verifying the average water temperature at the Unit 3 intake structure to be within limits when the average water temperature exceeds 70°F.

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6 Flood protection shall be provided for the service water pump cubicles and components when the water level exceeds 13 feet Mean Sea Level, USGS datum, at the Unit 3 intake structure.

APPLICABILITY: At all times.

ACTION:

With the water level at 13 feet above Mean Sea Level, USGS datum, at the Unit 3 intake structure, shut the watertight doors of both service water pump cubicles within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.7.6 The water level at the Unit 3 intake structure shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 8 feet above Mean Sea Level, USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation 8 feet above Mean Sea Level, USGS datum.

PLANT SYSTEMS

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Control Room Emergency Air Filtration Systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Emergency Air Filtration 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 Emergency Air Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency Air Filtration System in the recirculation mode.

b. With both Control Room Emergency Air Filtration Systems inoperable, or with the OPERABLE Control Room Emergency Air Filtration System required to be in the recirculation mode by ACTION a. not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes

SURVEILLANCE REQUIREMENTS

4.7.7 Each Control Room Emergency Air Filtration 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 95°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 charcoal adsorbers and verifying the system flowrate against the system pressure drop using certified fan curves and 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 charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% 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 system flow rate is 1120 cfm ± 20%;

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 0.175%; and

3) Verifying a system flow rate of 1120 cfm ± 20% during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978,* for a methyl iodide penetration of less than 0.175%;

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.75 inches Water Gauge while operating the system at a flow rate of 1120 cfm ± 20% and verifying the fan curve for observed pressure drop against the system flowrate;

2) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 230 cfm relative to adjacent areas during system operation; and

3) Verifying that the heaters dissipate 9.4 ± 1 kW when tested in accordance with ANSI N510-1980.

SURVEILLANCE REQUIREMENTS (Continued)

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 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1120 cfm ± 20%; and

g. 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 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 1120 cfm ± 20%.

*ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent Control Room Envelope Pressurization Systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

a. With one Control Room Envelope Pressurization System inoperable either:

   1. Restore the inoperable system to OPERABLE status within 7 days, or

   2. Initiate and maintain operation of an OPERABLE Control Room Emergency Air Filtration System in the recirculation mode, or

   3. Be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours and suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both Control Room Envelope Pressurization Systems inoperable, within one hour initiate action to restore one inoperable system to OPERABLE status and either:

   1. Initiate and maintain operation of an OPERABLE Control Room Emergency Air Filtration System in the recirculation mode, or

   2. Be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours and suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.8 Each Control Room Envelope Pressurization System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the storage air bottles are pressurized to greater than or equal to 220 psig,

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

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or following a major alteration of the control room envelope pressure boundary by:

   1. Verifying that the control room envelope is isolated in response to a Control Building Isolation test signal,

   2. Verifying that after a 60 second time delay following a Control Building Isolation test signal, the control room envelope pressurizes to greater than or equal to 1/8 inch W.G. relative to the outside atmosphere, and

   3. Verifying that the positive pressure of Specification 4.7.8.c.2 is maintained for greater than or equal to 60 minutes.

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 Two independent Auxiliary Building Filter Systems shall be OPERABLE.

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

ACTION:

With one Auxiliary Building Filter 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

4.7.9 Each Auxiliary Building Filter 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 charcoal adsorbers and verifying the system flowrate against the system pressure drop using certified fan curves and 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 0.05% 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 0.175%; and

SURVEILLANCE REQUIREMENTS (Continued)

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 0.175%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.8 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ± 10% and verifying the fan curve for observed pressure drop against the system flowrate,

2) Verifying that the system starts on a Safety Injection test signal, and

3) Verifying that the heaters dissipate 180 ± 18 kW when tested in accordance with ANSI N510-1980.

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 0.05% 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 0.05% 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%.

*ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 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 on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.10g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and 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. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. 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 snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection shall be performed at the first refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type per Inspection Period	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 irrespective 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 4.7.10f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least 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.

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 sample plan for each type 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 the total of each type of snubber 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 4.7.10f., an additional 5% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

e. Functional Tests (Continued)

2) A representative sample of each type of snubber 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 4.7.10f. The cumulative number of snubbers of a type 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 of that type of snubber 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.

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 the snubbers of each type 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 of each type. 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.

SURVEILLANCE REQUIREMENTS (Continued)

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;

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.

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 4.7.10e. 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 life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.3.

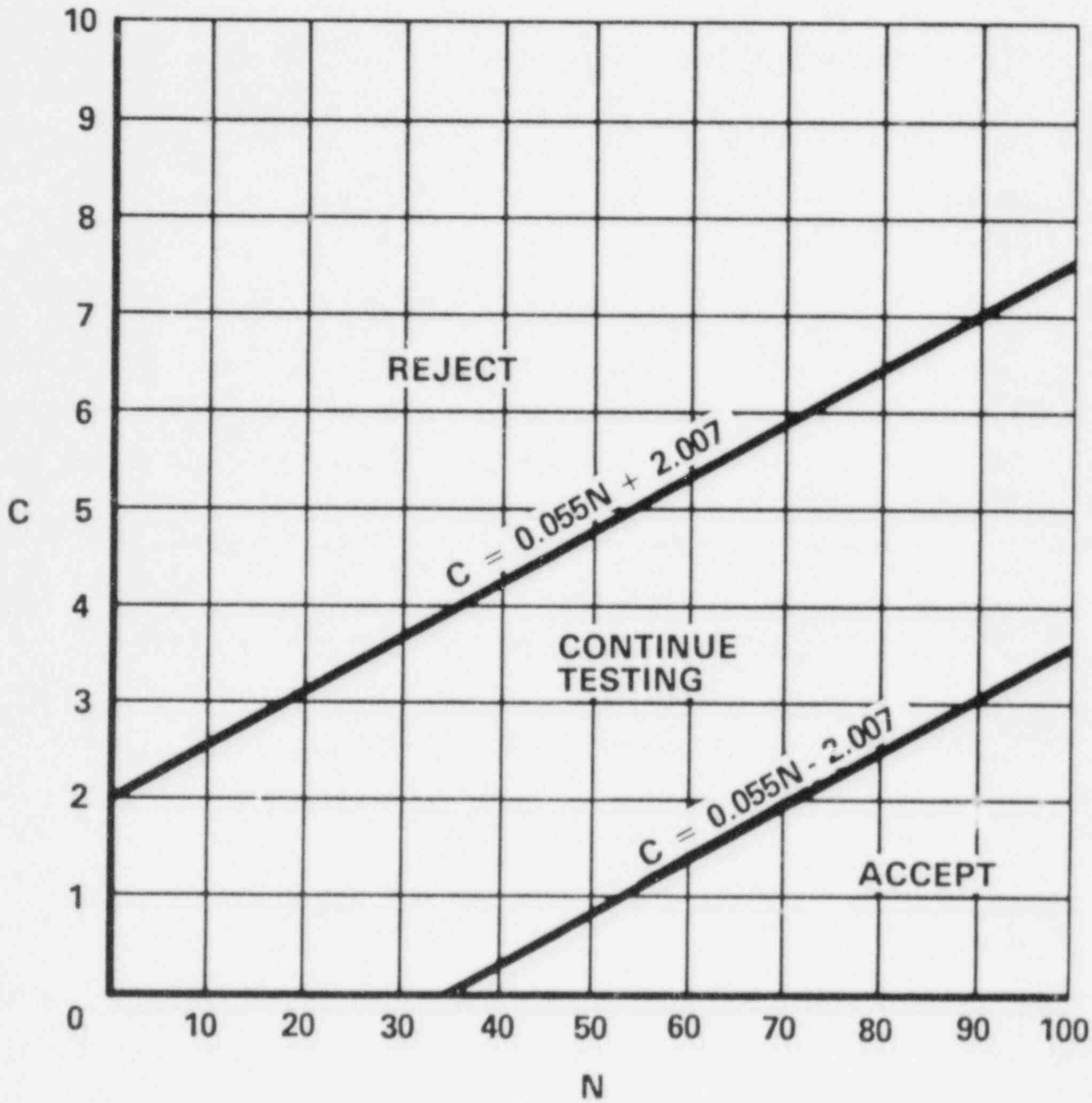


FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

3/4.7.11 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.11 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

4.7.11.1 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.

4.7.11.2 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.

4.7.11.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

3/4.7.12 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.12.1 The Fire Suppression Water System shall be OPERABLE with:

a. At least three fire suppression pumps, each with a capacity of 1800 gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies, each with a minimum contained volume of 200,000 gallons, and

c. An OPERABLE flow path capable of taking suction from the fire water tanks and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, hose standpipes, the first valve upstream of the water flow alarm device on each sprinkler and the first valve upstream of the deluge valve on each Deluge or Spray System required to be OPERABLE per Specifications 3.7.12.2, 3.7.12.5, and 3.7.12.6.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply 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 two pumps inoperable, establish a continuous fire watch of the turbine building with back-up fire suppression equipment within 1 hour. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

c. With the Fire Suppression Water System otherwise inoperable, establish a backup Fire Suppression Water System within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.12.1.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume,

b. 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,

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) outside containment in the flow path is in its correct position,

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) inside containment in the flow path 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 pump develops at least 1800 gpm at a system head of 227 feet,

   2) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   3) Verifying that each fire suppression pump starts sequentially to maintain the Fire Suppression Water System pressure greater than or equal to 75 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.12.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel storage tank contains at least 125 gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 20 minutes.

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-D270-1965 is within the acceptable limits specified in Table 1 of ASTM D975-1974 when checked for viscosity and water and sediment; and

c. At least once per 18 months by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.12.1.3 The fire pump diesel starting 12-volt batteries and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The electrolyte level of each battery cell is above the plates, and

2) The voltage of each battery is greater than or equal to 12 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the batteries, 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, free of corrosion, and coated with anticorrosion material.

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.12.2 The following Deluge Spray and/or Sprinkler Systems shall be OPERABLE:

a. A Emergency Generator Enclosure Sprinkler,

b. B Emergency Generator Enclosure Sprinkler,

c. A RSST Deluge,

d. B RSST Deluge,

e. A Fuel Building Filter Bank Deluge,

f. B Fuel Building Filter Bank Deluge,

g. A Auxiliary Building Filter Bank Deluge,

h. B Auxiliary Building Filter Bank Deluge,

i. A Supplementary Leak Collection Filter Bank Deluge,

j. B Supplementary Leak Collection Filter Bank Deluge,

k. Containment Cable Penetration Area Sprinkler,

l. Charging Pump Water Curtain Sprinkler System, and

m. ESF Building Water Curtain Sprinkler System.

APPLICABILITY: Whenever equipment protected by the Deluge Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Deluge 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

4.7.12.2 Each of the above required Deluge 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) outside containment in the flow path 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) inside containment in the flow path is in its correct position,

d. At least once per 18 months:

   1) By performing a system functional test which includes simulated automatic actuation of the system, and:

      a) Verifying that the deluge valves in the flow path actuate to their correct positions on a simulated test signal, and

      b) 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 the dry pipe deluge and sprinkler headers to verify their integrity; and

   3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.*

e. At least once per 3 years by performing an air flow or water test through each open head deluge header and verifying each open head deluge nozzle is unobstructed.*

---

*Not applicable to the Fuel Building, Auxiliary Building, or Supplementary Leak Collection Filter Banks.

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.12.3 The following CO_2 Systems shall be OPERABLE:

a. A Emergency Generator Fuel Oil Tank Vault,

b. B Emergency Generator Fuel Oil Tank Vault,

c. North Electrical Tunnel,

d. South Electrical Tunnel,

e. Cable Spreading Room,

f. West Switchgear Room,

g. East Switchgear Room,

h. A MCC and Rod Control Area, and

i. B MCC and Rod Control Area.

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

4.7.12.3.1 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.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.12.3.2 Each of the above required 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 50% of volume and pressure to be greater than 275 psig, and

b. At least once per 18 months by verifying:

   1) The system, including valves and associated ventilation system fire dampers, actuates manually and automatically upon receipt of a simulated actuation signal, and

   2) Flow from each nozzle during a "Puff Test."

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.12.4 The Instrument Rack Room Underfloor Area Halon System shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the Halon System is required to be OPERABLE.

ACTION:

a. With the above required Halon System 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

4.7.12.4 The above required Halon 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 is in its correct position,

b. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight and pressure to be at least 90% of full charge pressure, and

c. At least once per 18 months by:

   1) Verifying the system, including associated Ventilation System fire dampers, actuates manually and automatically, upon receipt of a simulated actuation signal, and

   2) Performance of a flow test through headers and nozzles to assure no blockage.

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.12.5 The fire hose stations given in Table 3.7-4 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 3.7-4 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 operating technicians, 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

4.7.12.5 Each of the fire hose stations given in Table 3.7-4 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 re-racking, 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 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION*	ELEVATION	HOSE RACK NUMBER
Containment	-24'6"	86, 90
Containment	3'8"	85, 89
Containment	24'6"	84, 88, 105
Containment	51'4"	83, 87
Auxiliary Building	4'6"	45 - 49
Auxiliary Building	24'6"	50 - 53
Auxiliary Building	43'6"	54 - 57
Auxiliary Building	66'6"	58 - 62
A Diesel Generator Enclosure	24'6"	78
B Diesel Generator Enclosure	24'6"	79
Fuel Building	11'0"	65
Fuel Building	24'6"	66 - 68
Fuel Building	52'4"	94

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.12.6 The yard fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses given in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.12.6 Each of the yard fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the hydrant hose house to assure all required equipment is at the hose house,

b. At least once per 6 months (once during March, April, or May and once during September, October, or November), by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged, and

c. At least once per 12 months by:

   1) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater,

   2) Inspecting all the gaskets and replacing any degraded gaskets in the couplings, and

   3) Performing a flow check of each hydrant to verify its OPERABILITY.

TABLE 3.7-5

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION*	HYDRANT NUMBER
West Yard Header	6, 7, 8, 9
North Yard Header	4, 5
East Yard Header	2, 3
South Yard Header	1, 10, 11

3/4.7.13 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.13 All fire rated assemblies (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 rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.13.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. A functional test of at least 10% of fire dampers installed in fire rated floor/wall assemblies. If functional testing acceptance criteria are not met, functional testing of an additional 10% of fire dampers will be performed. This testing process will continue until a 10% sample of fire dampers has been found to satisfy acceptance criteria. Functional testing of fire dampers will be performed so that a 100% verification of the operability of the fire dampers will be achieved 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 will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.13.2 Each of the above required fire doors shall be verified OPERABLE by inspecting closing mechanism and latches at least once per 6 months, and by verifying:

a. At least once per 7 days, each locked closed fire door is closed, and

b. At least once per 24 hours, each unlocked fire door is closed.

3/4.7.14 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.14 The temperature limit of each area shown in Table 3.7-6 shall not be exceeded.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6:

a. By less than 20°F and for less than 8 hours, record the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s).

b. By less than 20°F and for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

c. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by more than 20°F, prepare and submit a Special Report as required by ACTION b. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.14 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1. AUXILIARY BUILDING	
AB-02, VCT and Boric Acid Transfer Pump Area, El 43'6"	≤ 120
AB-03, Charging Pump Area, El 24'6"	≤ 110
AB-04, General Area, El 66'6"	≤ 120
AB-06, General Area, El 43'6"	≤ 120
AB-07, General Area, El 4'6"	≤ 120
AB-08, General Area (East), El 4'6"	≤ 120
AB-09, General Area (South), El 4'6"	≤ 120
AB-10, General Area, El 4'6"	≤ 120
AB-11, General Area, El 43'6"	≤ 120
AB-13, General Area (North), El 4'6"	≤ 120
AB-16, Supplemental Leak Collection Filter Area, El 66'6"	≤ 120
AB-19, MCC/Rod Drive Area, El 24'6"	≤ 120
AB-21, MCC Air Conditioning Room, El 66'6"	≤ 120
AB-22, Rod Drive Area, El 43'6"	≤ 120
AB-25, Charging Pump Area, El 24'6"	≤ 110
AB-26, RPCCW Pump Area, El 24'6"	≤ 110
AB-29, General Area (Southeast), El 24'6"	≤ 120
AB-33, Boric Acid Tank Area, El 43'6"	≤ 120
AB-35, Boric Acid Tank Area, El 43'6"	≤ 120
AB-39, Fuel Building and Auxiliary Building Filter Area, El 66'6"	≤ 120

TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
2. CONTROL BUILDING	
CB-01, Switchgear and Battery Rooms, El 4'6"	≤ 104
CB-02, Cable Spreading Room, El 24'6"	≤ 110
CB-03, Control and Computer Rooms, El 47'6"	≤ 95
CB-04, Chiller Room, El 64'6"	≤ 104
CB-05, Mechanical Equipment Room, El 64'6"	≤ 104
3. CONTAINMENT	
CS-01, Inside Crane Wall, El all	≤ 120
CS-02, Outside Crane Wall, El all	≤ 120
4. INTAKE STRUCTURE	
CW-01, Entire Building	≤ 110
5. DIESEL GENERATOR BUILDING	
DG-01, Entire Building	≤ 120
6. ESF BUILDING	
ES-01, HVAC and MCC Area, El 36'6"	≤ 110
ES-02, SIH Pump Area, El 21'6"	≤ 110
ES-03, Pipe Tunnel Area, El 4'6"	≤ 110
ES-04, RHS Cubicles, El all	≤ 110
ES-05, RSS Cubicles, El all	≤ 110
ES-06, Motor Driven Auxiliary Feedwater Pump Area, El 24'6"	≤ 110
ES-07, Turbine Driven Auxiliary Feedwater Pump Area, El 24'6"	≤ 110

TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
7. FUEL BUILDING	
FB-02, Fuel Pool Pump Cubicles, El 24'6"	≤ 110
FB-03, General Area, El 52'4"	≤ 104
8. FUEL OIL VAULT	
FV-01, Diesel Fuel Oil Vault	≤ 95
9. HYDROGEN RECOMBINER BUILDING	
HR-01, Recombiner Skid Area, El 24'6"	≤ 125
HR-02, Controls Area, El 24'6"	≤ 110
HR-03, Sampling Area, El 24'6"	≤ 110
HR-04, HVAC Area, El 37'6"	≤ 110
10. MAIN STEAM VALVE BUILDING	
MS-01, Entire Building	≤ 120
11. TURBINE BUILDING	
TB-01, Entire Building	≤ 115
12. TUNNEL	
TN-02, Pipe Tunnel-Auxiliary, Fuel and ESF Building	≤ 112
13. YARD	
YD-01, Yard	≤ 115

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System, and

b. Two separate and independent diesel generators, each with:

1) A separate day tank containing a minimum volume of 205 gallons of fuel,

2) A separate Fuel Storage System containing a minimum volume of 32,760 gallons of fuel,

3) A separate fuel transfer pump,

4) Lubricating oil storage containing a minimum total volume of 280 gallons of lubricating oil, and

5) Capability to transfer lubricating oil from storage to the diesel generator unit.

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.5) 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.5) 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, 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 the requirements of Specification 4.8.1.1.2a.5) 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 the requirements of 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 least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at 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 pump starts and transfers fuel from the storage system to the day tank,

   4) Verifying the lubricating oil inventory in storage,

   5) Verifying the diesel starts from ambient condition and accelerates to at least 508 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 0.8 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

      a) Manual, 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.

   6) Verifying the generator is synchronized, loaded to greater than or equal to 4986 kW in less than or equal to 60 seconds, and operates with a load greater than or equal to 4986 kW for at least 60 minutes, and

   7) 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 sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

   1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

---

*All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures (e.g., gradual acceleration and/or gradual loading >60 sec) as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS (Continued)

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 not 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.

e. 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;

f. 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 595 kW while maintaining voltage at 4160 $\pm$ 420 volts and frequency at 60 $\pm$ 3 Hz;

3) Verifying the generator capability to reject a load of 4986 kW without tripping. The generator voltage shall not exceed 4784 volts 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 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 0.8 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 4160 ± 420 volts and 60 ± 0.8 Hz within 10 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 10 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 ± 0.8 Hz during this test; and

c) Verifying that all automatic diesel generator trips, except engine overspeed, lube oil pressure low (2 of 3 logic) and generator differential, 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. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 5485 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 4986 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 0.8 Hz within 10 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 4.8.1.1.2f.6)b);*

8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 5335 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 pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval; and

13) Verifying that the following diesel generator lockout features prevent diesel generator starting:

a) Engine overspeed,

b) Lube oil pressure low (2 of 3 logic),

c) Generator differential, and

d) Emergency stop.

---

*If Specification 4.8.1.1.2f.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 4986 kW for 1 hour or until operating temperature has stabilized.

SURVEILLANCE REQUIREMENTS (Continued)

g. 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 508 rpm in less than or equal to 10 seconds; and

h. 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, 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.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission in a Special Report 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.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
≤ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
≥ 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 the 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."

LIMITING CONDITION FOR OPERATION

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

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

   1) A day tank containing a minimum volume of 205 gallons of fuel,

   2) A fuel storage system containing a minimum volume of 32,760 gallons of fuel,

   3) A fuel transfer pump,

   4) Lubricating oil storage containing a minimum total volume of 280 gallons of lubricating oil, and

   5) Capability to transfer lubricating oil from storage to the diesel generator unit.

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 a greater than or equal to 7.0 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.6)), and 4.8.1.1.3.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery Bank 301A-1, and an associated full capacity charger,

b. 125-volt Battery Bank 301A-2, and an associated full capacity charger,

c. 125-volt Battery Bank 301B-1 and an associated full capacity charger, and

d. 125-volt Battery Bank 301B-2 and an associated full capacity charger.

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

ACTION:

a. With either Battery Bank 301A-1 or 301B-1, and/or one of the required full capacity chargers inoperable, restore the inoperable battery bank and/or full capacity charger 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 either Battery Bank 301A-2 or 301B-2 inoperable, restore the inoperable battery bank 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

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2a meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 129 volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

   1) The parameters in Table 4.8-2a 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, 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) Each battery charger will supply at least the amperage indicated in Table 4.8-2b at 125 volts for at least 24 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; 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.

TABLE 4.8-2a

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 < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(6)	> 2.07 volts
Specific Gravity(4)	≥ 1.200(5)	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 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.

TABLE 4.8-2b

BATTERY CHARGER CAPACITY

CHARGER	AMPERAGE
301A-1	200
301A-2	50
301A-3	200
301B-1	200
301B-2	50
301B-3	200

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt battery bank and its associated full-capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required battery bank and/or full-capacity charger 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 a 7.0 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery bank and full-capacity charger shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner:

a. Train A A.C. Emergency Busses consisting of:

1) 4160-Volt Emergency Bus #34C, and
2) 480-Volt Emergency Bus #32R, 32S, 32T, and 32Y.

b. Train B A.C. Emergency Busses consisting of:

1) 4160-Volt Emergency Bus #34D, and
2) 480-Volt Emergency Bus #32U, 32V, 32W, and 32X.

c. 120-Volt A.C. Vital Bus #VIAC-1 energized from its associated inverter connected to D.C. Bus #301A-1*,

d. 120-Volt A.C. Vital Bus #VIAC-2 energized from its associated inverter connected to D.C. Bus #301B-1*,

e. 120-Volt A.C. Vital Bus #VIAC-3 energized from its associated inverter connected to D.C. Bus #301A-2*,

f. 120-Volt A.C. Vital Bus #VIAC-4 energized from its associated inverter connected to D.C. Bus #301B-2*,

g. 125-Volt D.C. Bus #301A-1 energized from Battery Bank #301A-1,

h. 125-Volt D.C. Bus #301A-2 energized from Battery Bank #301A-2,

i. 125-Volt D.C. Bus #301B-1 energized from Battery Bank #301B-1, and

j. 125-Volt D.C. Bus #301B-2 energized from Battery Bank #301B-2.

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

ACTION:

a. With one of the required trains of A.C. emergency busses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN

*Two inverters may be disconnected from their D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

within the following 30 hours; and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. emergency busses consisting of one 4160-volt and four 480-volt A.C. emergency busses,

b. Two 120-volt A.C. vital busses energized from their associated inverters connected to their respective D.C. busses, and

c. Two 125-volt D.C. busses energized from their associated battery banks.

APPLICABILITY MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible, and within 8 hours, depressurize and vent the RCS through at least a 7.0 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 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 Table 3.8-1 inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 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 system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed, 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 with a value equal to 300% of the pickup of the long-time delay trip element and 150% of the pickup of the short-time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Device Number	System Powered
3RCS-HTR-24	Reactor Coolant System
3RCS-HTR-51	Reactor Coolant System
3RCS-HTR-52	Reactor Coolant System
3RCS-HTR-29	Reactor Coolant System
3RCS-HTR-57	Reactor Coolant System
3RCS-HTR-58	Reactor Coolant System
3RCS-HTR-34	Reactor Coolant System
3RCS-HTR-63	Reactor Coolant System
3RCS-HTR-64	Reactor Coolant System
3RCS-HTR-39	Reactor Coolant System
3RCS-HTR-69	Reactor Coolant System
3RCS-HTR-70	Reactor Coolant System
3RCS-HTR-44	Reactor Coolant System
3RCS-HTR-75	Reactor Coolant System
3RCS-HTR-76	Reactor Coolant System
3RCS-HTR-23	Reactor Coolant System
3RCS-HTR-49	Reactor Coolant System
3RCS-HTR-50	Reactor Coolant System
3RCS-HTR-28	Reactor Coolant System
3RCS-HTR-55	Reactor Coolant System
3RCS-HTR-56	Reactor Coolant System
3RCS-HTR-33	Reactor Coolant System
3RCS-HTR-61	Reactor Coolant System
3RCS-HTR-62	Reactor Coolant System

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Device Number	System Powered
3RCS-HTR-38	Reactor Coolant System
3RCS-HTR-67	Reactor Coolant System
3RCS-HTR-68	Reactor Coolant System
3RCS-HTR-43	Reactor Coolant System
3RCS-HTR-73	Reactor Coolant System
3RCS-HTR-74	Reactor Coolant System
3CCP*MOV226	Reactor Plant Component Cooling Water System
3CCP*MOV227	Reactor Plant Component Cooling Water System
3CCP*MOV228	Reactor Plant Component Cooling Water System
3CCP*MOV229	Reactor Plant Component Cooling Water System
3CCP*MOV48B	Reactor Plant Component Cooling Water System
3CMS*MOV24	Containment Atmosphere Monitoring System
3RCS*MV8000A	Reactor Coolant System
3CCP*MOV222	Reactor Plant Component Cooling Water System
3CCP*MOV223	Reactor Plant Component Cooling Water System
3CCP*MOV224	Reactor Plant Component Cooling Water System
3CCP*MOV225	Reactor Plant Component Cooling Water System
3CCP*MOV48A	Reactor Plant Component Cooling Water System
3CHS*MV8112	Chemical and Volume Control System
3RCS*MV8000B	Reactor Coolant System
3RCS*MV8098	Reactor Coolant System
3RHS*MV8702B	Residual Heat Removal System
3RHS*MV8702C	Residual Heat Removal System
3SIL*MV8808B	Low Pressure Safety Injection System

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Device Number	System Powered
3SIL*MV8808D	Low Pressure Safety Injection System
3RHS*MV8701A	Residual Heat Removal System
3RHS*MV8701C	Residual Heat Removal System
3SIL*MV8808A	Low Pressure Safety Injection System
3SIL*MV8808C	Low Pressure Safety Injection System
3RCS*MV8001B	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8001D	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8002A	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8002B	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8001C	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8002D	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8003B	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8003D	Reactor Coolant System (Limit Switch Comp. Htr)
3RMS*RM42	Radiation Monitoring System
3RCS*MV8001A	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8002C	Reactor Coolant System (Limit Switch Comp. Htr)
3RCS*MV8003C	Reactor Coolant System (Limit Switch Comp. Htr)
3RMS*RM41	Radiation Monitoring System
3JB-0025	Junction Box
HTR-11, 12, 35	Pressurizer Heater System
HTR-15, 16, 40	Pressurizer Heater System
HTR-19, 20, 45	Pressurizer Heater System
3JB-0026	Junction Box
HTR-3, 4, 25	Pressurizer Heater System
HTR-7, 8, 30	Pressurizer Heater System

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Device Number	System Powered
3JB-0025	Junction Box
HTR-13, 14, 37	Pressurizer Heater System
HTR-17, 18, 42	Pressurizer Heater System
3JB-0026	Junction Box
HTR-1, 2, 22	Pressurizer Heater System
HTR-5, 6, 27	Pressurizer Heater System
HTR-9, 10, 32	Pressurizer Heater System
3HVU-FN2C	Containment Structure Ventilation System
3DGS-P1A	Reactor Plant Gaseous Drains System
3DGS-P1B	Reactor Plant Gaseous Drains System
3DGS-P3A	Reactor Plant Gaseous Drains System
3DGS-P3B	Reactor Plant Gaseous Drains System
3LAR-PNL3RC1O	Lighting in Reactor Building System
3LAR-PNL3RC2P	Lighting in Reactor Building System
3JRB-EL	Reactor Building Superstructure Loads
3MHR-CRN2	Material Handling Crane in Reactor Building
3CVS*MOV25	Containment Vacuum System
3RCS*MV8001A	Reactor Coolant System
3RCS*MV8001C	Reactor Coolant System
3RCS*MV8002A	Reactor Coolant System
3RCS*MV8002C	Reactor Coolant System
3RCS*MV8003A	Reactor Coolant System
3RCS*MV8003C	Reactor Coolant System
3RCS-P1A1	Reactor Coolant System
3DAS-P2A	Reactor Plant Aerated Drains System

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Device Number	System Powered
3RCS-P1C1	Reactor Coolant System
3HVU-FN1C	Containment Structure Ventilation System
3RCS-TBP1A-2	Reactor Coolant System
3RCS-TBP1C-2	Reactor Coolant System
3FNT-CONTR	Fuel Nuclear Transfer System
3MHR-CRN2	Material Handling Crane in Reactor Building
3SES-TL17A	Nuclear Servicing Equipment System
3SES-TL17C	Nuclear Servicing Equipment System
3HVU-PNL-01A	Containment Structure Ventilation System
3HVU-PNL-01B	Containment Structure Ventilation System
3LAR-XLR1	Lighting in Reactor Building System
3LAR-XLR2	Lighting in Reactor Building System
3LAR-XLR3	Lighting in Reactor Building System
3LAR-XLR4	Lighting in Reactor Building System
3HVU-FN2A	Containment Structure Ventilation System
3HVU-FN2B	Containment Structure Ventilation System
3JB-0025	Junction Box
HTR-21, 47, 48	Pressurizer Heater System
HTR-26, 53, 54	Pressurizer Heater System
HTR-31, 59, 60	Pressurizer Heater System
3JB-0026	Junction Box
HTR-36, 65, 66	Pressurizer Heater System
HTR-41, 71, 72	Pressurizer Heater System
HTR-46, 77, 78	Pressurizer Heater System
3DAS-P1	Reactor Plant Aerated Drains System
3DAS-P10	Reactor Plant Aerated Drains System

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Device Number	System Powered
3RCS-TBP1B-2	Reactor Coolant System
3RCS-TBP1D-2	Reactor Coolant System
3RDI-CABA	Reactor Rod Drive Instrumentation System
3RDI-CABB	Reactor Rod Drive Instrumentation System
3NMT-TB1	Neutron Monitoring Traverse Incore Probe System (Associated with Controller 3NMT-CONT 1A7)
3RCS-P1B1	Reactor Coolant System
3RCS-P1D1	Reactor Coolant System
3DAS-P2B	Reactor Plant Aerated Drains System
3RCS*MV8001B	Reactor Coolant System
3RCS*MV8001D	Reactor Coolant System
3RCS*MV8002B	Reactor Coolant System
3RCS*MV8002D	Reactor Coolant System
3RCS*MV8003D	Reactor Coolant System
3RCS*MV8003B	Reactor Coolant System
3SES-TL17B	Nuclear Servicing Equipment System
3HVU-FLT1A	Containment Structure Ventilation System
3POP-JB1R1	Power Outlets General Purpose 3 Phase System
3POP-JB2R1	Power Outlets General Purpose 3 Phase System
3HVU-FN3A	Containment Structure Ventilation System
3IAC-C1A	Containment Instrument Air System
3FNT-CONTR	Fuel Nuclear Transfer System (Control Power)
3RMS-RM01	Radiation Monitoring System
3RMS-RM02	Radiation Monitoring System

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Device Number	System Powered
3RMS-RM03	Radiation Monitoring System
3RMS-RM31	Radiation Monitoring System
3RMS-RM32	Radiation Monitoring System
3RMS-RM35	Radiation Monitoring System
3IAC-DRY1	Containment Instrument Air System
3HVU-FS34A,B,C	Containment Structure Ventilation System
3HVU-PDIS20A, 21A, 22A, 23A	Containment Structure Ventilation System
3HVU-PDIS 20B, 21B, 22B, 23B	Containment Structure Ventilation System
3HVU-FN2C	Containment Structure Ventilation System (Motor Heater)
3HVU-FS36A,B,C,D	Containment Structure Ventilation System
3COP-AMP200, 201, 202, 203, 220, 221, 240, 241, 260, 261, 262	Communication Paging System
3HVU-FN1B	Containment Structure Ventilation System (Motor Heater)
3HVU-FN1A	Containment Structure Ventilation System
3HVU-FN1C	Containment Structure Ventilation System (Motor Heater)
3HVU-FLT1B	Containment Structure Ventilation System
3HVU-FN3B	Containment Structure Ventilation System
3IAC-C1B	Containment Instrument Air System
3POP-JB3R1	Power Outlets General Purpose 3 Phase System
3MHR-CRN1	Material Handling Crane in Reactor Building
3RCS-P1C	Reactor Coolant System
3RCS-P1B	Reactor Coolant System
3RCS-P1A	Reactor Coolant System

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Device Number	System Powered
3RCS-P1D	Reactor Coolant System
3DAS-JB01	Aerated Drains Junction Box
3MHR-CRN-3A	Material Handling Cranes in Containment
3MHR-CRN-3B	Material Handling Cranes in Containment
3MHR-CRN-3C	Material Handling Cranes in Containment
3MHR-CRN-3D	Material Handling Cranes in Containment
3MHR-CRN-4	Material Handling Cranes in Containment
3MHR-CRN-5	Material Handling Cranes in Containment
3HVU-FN 1B	Containment Structure Ventilation System
3FNT-CONTR	Fuel Nuclear Transfer System (Heater)
3NMT-TB1	Neutron Monitoring Traverse Incore Probe System (Associated with Controller 3NMT-CONT3A7)
3ERS-NBY23	Seismic Instrument Sensor
3ECS-TT71	Containment Structure Temperature Trans.
3SIL*MOV8808B	Low Pressure Safety Injection System (Valve Position Indication)
3SIL*MOV8808D	Low Pressure Safety Injection System (Valve Position Indication)
3IAS*MOV72	Instrument Air System
3SIL*MOV8808A	Low Pressure Safety Injection System (Valve Position Indication)
3HVU-FN2B	Containment Structure Ventilation System (Motor Heater)
3HVU-FN2A	Containment Structure Ventilation System (Motor Heater)
3HVU-FN1A	Containment Structure Ventilation System (Motor Heater)
3SIL*MOV8808C	Low Pressure Safety Injection System (Valve Position Indication)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2.1 The thermal overload protection of each valve given in Table 3.8-2a shall be bypassed only under accident conditions by an OPERABLE bypass device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed by the appropriate accident signal(s) by performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry during COLD SHUTDOWN or REFUELING at least once per 18 months.

TABLE 3.8-2a

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASSED ONLY UNDER ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3CCP*MOV45A	Containment Isolation Valve
3CCP*MOV45B	Containment Isolation Valve
3CCP*MOV48A	Containment Isolation Valve
3CCP*MOV48B	Containment Isolation Valve
3CCP*MOV49A	Containment Isolation Valve
3CCP*MOV49B	Containment Isolation Valve
3CCP*MOV222	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV223	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV224	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV225	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV226	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV227	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV228	Containment Air Recirculation Cooling Coil Supply
3CCP*MOV229	Containment Air Recirculation Cooling Coil Supply
3CHS*LCV112B	Volume Control Tank Outlet Isolation
3CHS*LCV112C	Volume Control Tank Outlet Isolation
3CHS*LCV112D	Volume Control Tank Outlet Isolation
3CHS*LCV112E	Volume Control Tank Outlet Isolation
3CHS*MV8100	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8105	Charging Pump to Reactor Coolant Isolation
3CHS*MV8106	Charging Pump to Reactor Coolant Isolation
3CHS*MV8110	Charging Pump Miniflow Isolation
3CHS*MV8111A	Charging Pump Miniflow Isolation

TABLE 3.8-2a (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
BYPASSED ONLY UNDER ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3CHS*MV8111B	Charging Pump Isolation
3CHS*MV8111C	Charging Pump Isolation
3CHS*MV8112	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8511A	Charging Pump Miniflow Control
3CHS*MV8511B	Charging Pump Miniflow Control
3CMS*MOV24	Containment Atmosphere Monitoring Inside Containment Isolation
3RHS*MV8701A	Residual Heat System Inlet Isolation
3RHS*MV8701B	Residual Heat System Inlet Isolation
3RHS*MV8702A	Residual Heat System Inlet Isolation
3RHS*MV8702B	Residual Heat System Inlet Isolation
3RSS*MOV20A	Containment Recirculation Spray Header Isolation
3RSS*MOV20B	Containment Recirculation Spray Header Isolation
3RSS*MOV20C	Containment Recirculation Spray Header Isolation
3RSS*MOV20D	Containment Recirculation Spray Header Isolation
3RSS*MOV23A	Containment Recirculation Pump Suction
3RSS*MOV23B	Containment Recirculation Pump Suction
3RSS*MOV23C	Containment Recirculation Pump Suction
3RSS*MOV23D	Containment Recirculation Pump Suction
3SIH*MV8801A	Charging Pump to Cold Leg Isolation
3SIH*MV8801B	Charging Pump to Cold Leg Isolation
3SIL*MV8808A	Accumulation Isolation
3SIL*MV8808B	Accumulation Isolation
3SIL*MV8808C	Accumulation Isolation

TABLE 3.8-2a (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
BYPASSED ONLY UNDER ACCIDENT CONDITIONS

VALVE NUMBER	SYSTEM(S) AFFECTED
3SIL*MV8808D	Accumulation Isolation
3SWP*MOV50A	Reactor Plant Component Cooling Water Heat Exchanger
3SWP*MOV50B	Reactor Plant Component Cooling Water Heat Exchanger
3SWP*MOV54A	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV54B	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV54C	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV54D	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV71A	Turbine Plant Component Cooling Water Heat Exchanger
3SWP*MOV71B	Turbine Plant Component Cooling Water Heat Exchanger
3SWP*MOV115A	Circulating Water Pump Bearing Lube Water Supply
3SWP*MOV115B	Circulation Water Pump Bearing Lube Water Supply
3SWP*MOV130A	Motor Control Center and Rod Control Service Water Isolation
3SWP*MOV130B	Motor Control Center and Rod Control Service Water Isolation
3IAS*MOV72	Containment Instrument Air Isolation
3QSS*MOV29A	Refueling Water Chemical Addition Tank Discharge
3QSS*MOV29B	Refueling Water Chemical Addition Tank Discharge

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION NOT BYPASSED

LIMITING CONDITION FOR OPERATION

3.8.4.2.2 The thermal overload protection of each valve given in Table 3.8-2b shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves inoperable, bypass the inoperable thermal overload within 8 hours; restore the inoperable thermal overload to OPERABLE status within 30 days or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.2 The thermal overload protection for the above required valves shall be demonstrated OPERABLE at least once per 18 months and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

TABLE 3.8-2b

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION NOT BYPASSED

VALVE NUMBER	SYSTEM(S) AFFECTED
3LMS*MOV40A	Containment Open Pressure Tap Isolation
3LMS*MOV40B	Containment Open Pressure Tap Isolation
3LMS*MOV40C	Containment Open Pressure Tap Isolation
3LMS*MOV40D	Containment Open Pressure Tap Isolation
3MSS*MOV17A	Steam Generator Auxiliary Feedwater Pump Steam Supply Non-Return
3MSS*MOV17B	Steam Generator Auxiliary Feedwater Pump Steam Supply Non-Return
3MSS*MOV17D	Steam Generator Auxiliary Feedwater Pump Steam Supply Non-Return
3QSS*MOV34A	Quench Spray Isolation
3QSS*MOV34B	Quench Spray Isolation
3SIH*MV8807B	Safety Injection High and Connection to Low Pressure Safety Injection
3SIH*MV8802B	Safety Injection Pump to Hot Leg Isolation
3CHS*MV8104	Boric Acid Filter to Charging Pump Isolation
3CHS*MV8109A	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8109B	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8109C	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8109D	Reactor Coolant Pump Seal Water Isolation
3CHS*MV8116	Charging Header Isolation
3CHS*MV8438A	Charging Header Isolation
3CHS*MV8438B	Charging Header Isolation
3CHS*MV8438C	Charging Header Isolation

TABLE 3.8-2b (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION
NOT BYPASSED

VALVE NUMBER	SYSTEM(S) AFFECTED
3CHS*MV8468A	Charging Pump Suction Isolation
3CHS*MV8468B	Charging Pump Suction Isolation
3CHS*MV8507A	Boric Acid Gravity Feed
3CHS*MV8507B	Boric Acid Gravity Feed
3FWA*MOV35A	Steam Generator Auxiliary Feedwater Pump Isolation
3FWA*MOV35B	Steam Generator Auxiliary Feedwater Pump Isolation
3FWA*MOV35C	Steam Generator Auxiliary Feedwater Pump Isolation
3FWA*MOV35D	Steam Generator Auxiliary Feedwater Pump Isolation
3RCS*MV8000A	Pressurizer Relief Isolation
3RCS*MV8000B	Pressurizer Relief Isolation
3RCS*MV8098	Reactor to Excess Letdown Isolation
3RHS*FCV610	Residual Heat Removal Pump Miniflow
3RHS*FCV611	Residual Heat Removal Pump Miniflow
3RHS*MV8701C	Residual Heat System Inlet Isolation
3RHS*MV8702C	Residual Heat System Inlet Isolation
3RHS*MV8716A	Residual Heat System Conn Isolation
3RHS*MV8716B	Residual Heat System Conn Isolation
3RSS*MOV38A	Containment Recirculation Pump Miniflow
3RSS*MOV38B	Containment Recirculation Pump Miniflow
3RSS*MV8837A	Recirculation Spray System to Residual Heat Removal System Cross Connection
3RSS*MV8837B	Recirculation Spray System to Residual Heat Removal System Cross Connection
3RSS*MV8838A	Recirculation Spray System to Residual Heat Removal System Cross Connection

TABLE 3.8-2b (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION NOT BYPASSED

VALVE NUMBER	SYSTEM(S) AFFECTED
3RSS*MV8838B	Recirculation Spray System to Residual Heat Removal System Cross Connection
3SIH*MV8802A	Safety Injection Pump to Hot Leg Isolation
3SIH*MV8806	Refueling Water Storage Tank to Safety Injection Isolation
3SIH*MV8807A	Safety Injection High & Connection to Low Pressure Safety Injection
3SIH*MV8813	High Pressure Safety Injection Pump Miniflow
3SIH*MV8814	High Pressure Safety Injection Pump Miniflow
3SIH*MV8821A	Safety Injection Pump Discharge Isolation
3SIH*MV8821B	Safety Injection Pump Discharge Isolation
3SIH*MV8835	SIH Supply to LPSI
3SIH*MV8920	High Pressure Safety Injection Pump Miniflow
3SIH*MV8923A	Safety Injection Pump Suction
3SIH*MV8923B	Safety Injection Pump Suction
3SIH*MV8924	Safety Injection/Charging Suction Cross Connection
3SIL*MV8804A	Residual Heat Removal Pump to Charging Pump
3SIL*MV8804B	Residual Heat Removal Pump to Charging Pump
3SIL*MV8809A	Residual Heat Removal Cold Leg Isolation
3SIL*MV8809B	Residual Heat Removal Cold Leg Isolation
3SIL*MV8812A	Refueling Water Storage Tank to Residual Heat Removal Pump
3SIL*MV8812B	Refueling Water Storage Tank to Residual Heat Removal Pump
3SIL*MV8840	Residual Heat Removal Hot Leg Isolation
3SWP*MOV24A	Service Water Backwash

TABLE 3.8-2b (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION NOT BYPASSED

VALVE NUMBER	SYSTEM(S) AFFECTED
3SWP*MOV24B	Service Water Backwash
3SWP*MOV24C	Service Water Backwash
3SWP*MOV24D	Service Water Backwash
3SWP*MOV57A	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV57B	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV57C	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV57D	Containment Recirculation Cooler Service Water Outlet
3SWP*MOV102A	Service Water Pump Discharge
3SWP*MOV102B	Service Water Pump Discharge
3SWP*MOV102C	Service Water Pump Discharge
3SWP*MOV102D	Service Water Pump Discharge
3MSS*MOV18A	Main Steam Pressure Relief Isolation
3MSS*MOV18B	Main Steam Pressure Relief Isolation
3MSS*MOV18C	Main Steam Pressure Relief Isolation
3MSS*MOV18D	Main Steam Pressure Relief Isolation
3MSS*MOV74A	Main Steam Pressure Relief Isolation
3MSS*MOV74B	Main Steam Pressure Relief Isolation
3MSS*MOV74C	Main Steam Pressure Relief Isolation
3MSS*MOV74D	Main Steam Pressure Relief Isolation
3CHS*MV8512A	Charging Pump Miniflow Control
3CHS*MV8512B	Charging Pump Miniflow Control

A.C. CIRCUITS INSIDE CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.3 At least the A.C. circuits for the following valves inside containment shall be de-energized:

Device Number	Valve
3SIL*MV8808A	Accumulator Isolation
3SIL*MV8808B	Accumulator Isolation
3SIL*MV8808C	Accumulator Isolation
3SIL*MV8808D	Accumulator Isolation

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.3 Each of the A.C. circuits for the above listed valves shall be determined to be de-energized at least once per 24 hours* by verifying that the associated circuit breakers are in the tripped condition.

*Except at least once per 31 days if locked, sealed or otherwise secured in the tripped condition.

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 33 gpm of a solution containing greater than or equal to 6300 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 Valve 3CHS-V305 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.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or 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 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 100 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 100 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 access 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) Be capable of being closed by an OPERABLE automatic containment purge and exhaust isolation valve.

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 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge and exhaust isolation valve within 100 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. Testing the containment purge and exhaust isolation valves per the applicable portions of Specification 4.6.3.2.

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 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine used for movement of fuel assemblies having:

1) A minimum capacity of 4000 pounds, and

2) An overload cutoff limit less than or equal to 3900 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

1) A minimum capacity of 3000 pounds, and

2) A load indicator which shall be used to prevent lifting loads in excess of 1000 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor 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 100 hours prior to the start of such operations by performing a load test of at least 4000 pounds and demonstrating an automatic load cutoff when the crane load exceeds 3900 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 100 hours prior to the start of such operations by performing a load test of at least 3000 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2200 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 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2200 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE or in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

---

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

---

*Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Purge and Exhaust Isolation System inoperable, close each of the purge and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere.

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

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge and exhaust isolation occurs on manual initiation and on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the 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

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 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

4.9.11 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.

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Building Exhaust Filter Systems shall be OPERABLE. At least one Fuel Building Exhaust Filter System shall be in operation whenever any evolution involving movement of fuel within the storage pool or crane operations with loads over the storage pool is in progress.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one Fuel Building Exhaust Filter System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Building Exhaust Filter System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Fuel Building Exhaust Filter System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Building Exhaust Filter System is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Building Exhaust Filter Systems 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 charcoal adsorbers and verifying the system flowrate against the system pressure drop using certified fan curves and 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:

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% 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 20,700 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 0.175%; and

3) Verifying a system flow rate of 20,700 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 0.175%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.8 inches Water Gauge while operating the system at a flow rate of 20,700 cfm ± 10% and verifying the fan curve for observed pressure drop against the system flowrate,

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation, and

3) Verifying that the heaters dissipate 150 ± 15 kW when tested in accordance with ANSI N510-1980.

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 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 20,700 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 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 20,700 cfm ± 10%.

4.9.12.2 The Fuel Building Exhaust Filter System shall be verified to be operating within 2 hours prior to the initiation of and at least once per 12 hours during either fuel movement within the fuel storage pool or crane operations with loads over the fuel storage pool.

*ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

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 33 gpm of a solution containing greater than or equal to 6300 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 33 gpm of a solution containing greater than or equal to 6300 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

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.10.2.1 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.1, 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 3.2.2.1 and 3.2.3.1 are maintained and determined at the frequencies specified in Specification 4.10.2.1.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2.1 or 3.2.3.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2.1 and 3.2.3.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1.1 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.

4.10.2.1.2 The Surveillance Requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.1.2 and 4.2.2.1.3, and

b. Specification 4.2.3.1.2.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

THREE LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.10.2.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.2, 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 55% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2.2 and 3.2.3.2 are maintained and determined at the frequencies specified in Specification 4.10.2.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2.2 or 3.2.3.2 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.2, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2.2 and 3.2.3.2, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.2.1 The THERMAL POWER shall be determined to be less than or equal to 55% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2.2 The Surveillance Requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2.2 and 4.2.2.2.3, and

b. Specification 4.2.3.2.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range 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.

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 Systems 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.

---

*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (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 from the site exceeding the above limits, restore the concentration to within the above limits within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program specified in Section I of the REMODCM.

4.11.1.1.2 The results of radioactive analyses shall be used in accordance with the methods of Section II of the REMODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

DOSE - LIQUIDS

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to any REAL MEMBER OF THE PUBLIC from radioactive materials in liquid effluents from Unit 3 released from the site (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 to be taken to reduce the releases of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose or dose commitment to any REAL MEMBER OF THE PUBLIC from such releases during the calendar year is within 3 mrem to the whole body and 10 mrem to any organ.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2.1 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with Section II of the REMODCM.

4.11.1.2.2 Relative accuracy or conservatism of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in the REMODCM.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate, at any time, offsite (see Figure 5.1-3) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be 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. The dose rate limit due to inhalation for Iodine-131, Iodine-133, tritium, and for all radioactive materials in particulate form with half-lives greater than 8 days shall be 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, decrease the release rate within 15 minutes to comply with the limit(s) given in Specification 3.11.2.1.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The release rate, at any time, of noble gases in gaseous effluents shall be controlled by the offsite dose rate as established above in Specification 3.11.2.1. The corresponding release rate shall be determined in accordance with the methodology of Section II of the REMODCM.

4.11.2.1.2 The noble gas effluent monitors of Specification 3.3.3.10 shall be used to control release rates to limit offsite doses within the values established in Specification 3.11.2.1.

4.11.2.1.3 The release rate of radioactive materials in gaseous effluents shall be determined by obtaining representative samples and performing analyses in accordance with the sampling and analysis program, specified in Section I of the REMODCM. The corresponding dose rate shall be determined using the methodology given in Section II of the REMODCM.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose offsite (see Figure 5.1-3) due to noble gases released from Unit 3 in gaseous effluents 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 to be taken to reduce the releases of radioactive noble gases in gaseous effluents during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose during the calendar year is within 10 mrad for gamma radiation and 20 mrad for beta radiation.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2.1 Dose Calculations. Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with Section II of the REMODCM once every 31 days.

4.11.2.2.2 Relative accuracy or conservatism of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in Section I of the REMODCM.

RADIOACTIVE EFFLUENTS

DOSE - RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to any REAL MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents from Unit 3 released offsite (see Figure 5.1-3) 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 radioiodines, radioactive materials in particulate form, or radionuclides other than 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 to be taken to reduce the releases during the remainder of the current calendar quarter and during the remainder of the calendar year so that the cumulative dose or dose commitment to any REAL MEMBER OF THE PUBLIC from such releases during the calendar year is within 15 mrem to any organ.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3.1 Dose Calculations. Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with Section II of the REMODCM once every 31 days.

4.11.2.3.2 Relative accuracy or conservatism of the calculations shall be confirmed by performance of the Radiological Environmental Monitoring Program as detailed in Section I of the REMODCM.

RADIOACTIVE EFFLUENTS

3/4.11.3 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.3 The dose or dose commitment to a REAL MEMBER OF THE PUBLIC from the Millstone site 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) over a period of 12 consecutive months.

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 3.11.1.2, 3.11.2.2, or 3.11.2.3, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 and limit the subsequent releases so that the dose or dose commitment to any REAL MEMBER OF THE PUBLIC from the Millstone site is limited to 25 mrem to the whole body or any organ (except thyroid, which is limited to 75 mrem) over 12 consecutive months. This Special Report shall include an analysis demonstrating that the radiation exposures to any REAL MEMBER OF THE PUBLIC from the Millstone site (including all effluent pathways and direct radiation) are less than 40 CFR Part 190 Standard. If the estimated dose(s) exceeds the above limits, 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

4.11.3 Dose Calculations. Cumulative dose contributions from liquid and gaseous effluents and direct radiation from the Millstone site shall be determined in accordance with Specifications 4.11.1.2.1, 4.11.2.2.1, and 4.11.2.3.1, and in accordance with Section II of the REMODCM once per 31 days.

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.

3/4.0 APPLICABILITY

BASES

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 The 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 Quench 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 Quench 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 Recirculation 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.

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 these 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.

BASES

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.

BASES

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, RCS boron concentration, and RCS 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 RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% $\Delta k/k$ 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. A 1.6% $\Delta k/k$ SHUTDOWN MARGIN is required to provide protection against a boron dilution accident.

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 -4.0×10^{-4} $\Delta k/k/^{\circ}F$. The MTC value of -3.1×10^{-4} $\Delta k/k/^{\circ}F$ 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 -4.0×10^{-4} $\Delta k/k/^{\circ}F$.

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 RCS 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 RCS 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 1.6% $\Delta k/k$ 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 21,020 gallons of 6300 ppm borated water from the boric acid storage tanks or 1,166,000 gallons of 2000 ppm borated water from the refueling water storage tank (RWST). A minimum RWST volume of 1,166,000 gallons is specified to be consistent with ECCS requirement.

With the RCS 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 3' F provides assurance that a mass addition pressure transient can be relie by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.6% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4100 gallons of 6300 ppm borated water from the boric acid storage tanks or 250,000 gallons of 2000 ppm borated water from the RWST.

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 RWST also ensure a pH value of between 7.0 and 7.5 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 minimum RWST solution temperature for MODES 5 and 6 is based on analysis assumptions in addition to freeze protection considerations. The minimum/maximum RWST solution temperatures for MODES 1, 2, 3 and 4 are based on analysis assumptions.

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

MOVABLE CONTROL ASSEMBLIES (Continued)

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 Indication 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.

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.

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

3/4.2 POWER DISTRIBUTION LIMITS

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 1.30 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 (four loops operating) or 2.60 (three loops operating) 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 Specifications 3.2.1.1 and 3.2.2.1 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 Figures 3.2-1a and 3.2-1b while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER (four loops operating) or between 32% and 65% of RATED THERMAL POWER (three loops operating). For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER (four loops operating) or between 15% and 32% of RATED THERMAL POWER (three loops operating), 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 two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During four loop 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. During three loop operation at THERMAL POWER levels between 32% and 65% and between 15% and 32% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figures B 3/4 2-1a and B 3/4 2-1b show typical monthly target bands.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS 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 ensure 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;

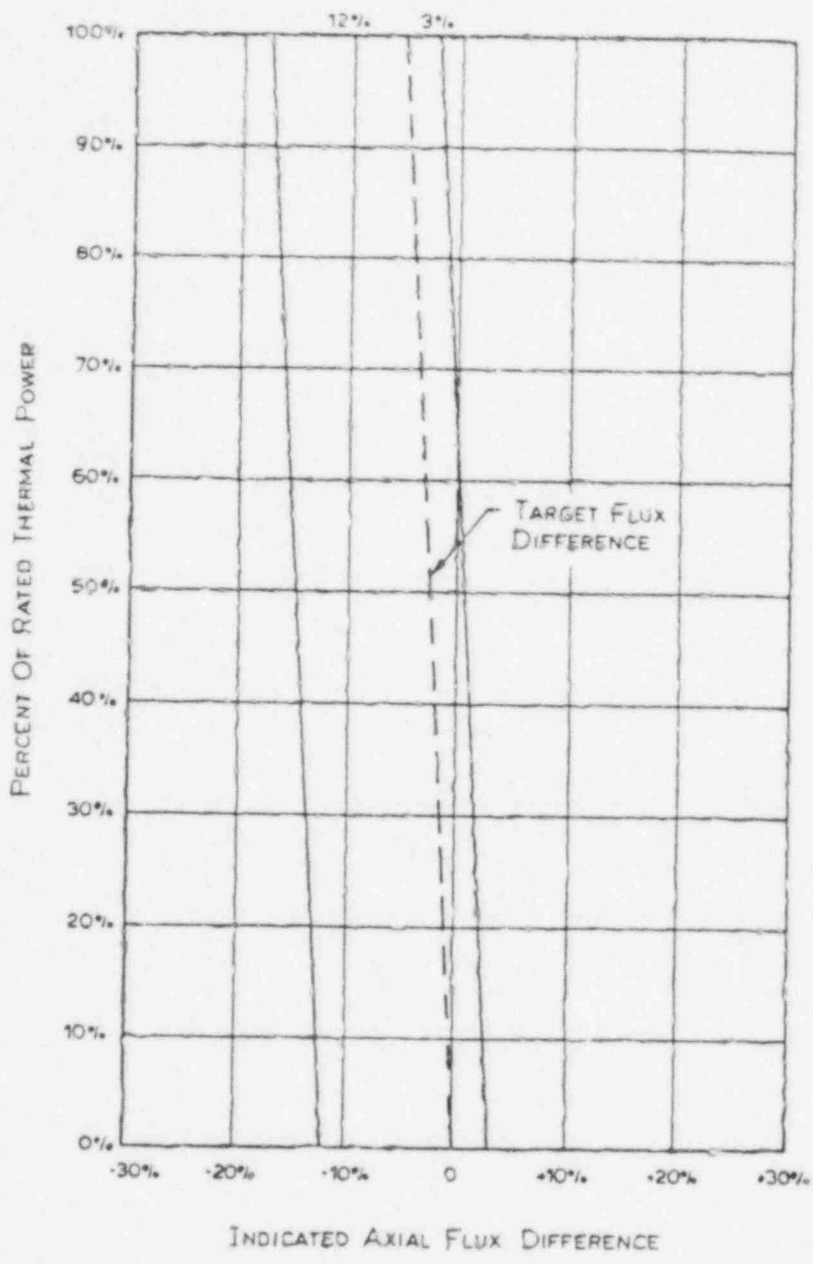


FIGURE B 3/4 2-1a
TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER
FOR FOUR LOOP OPERATION

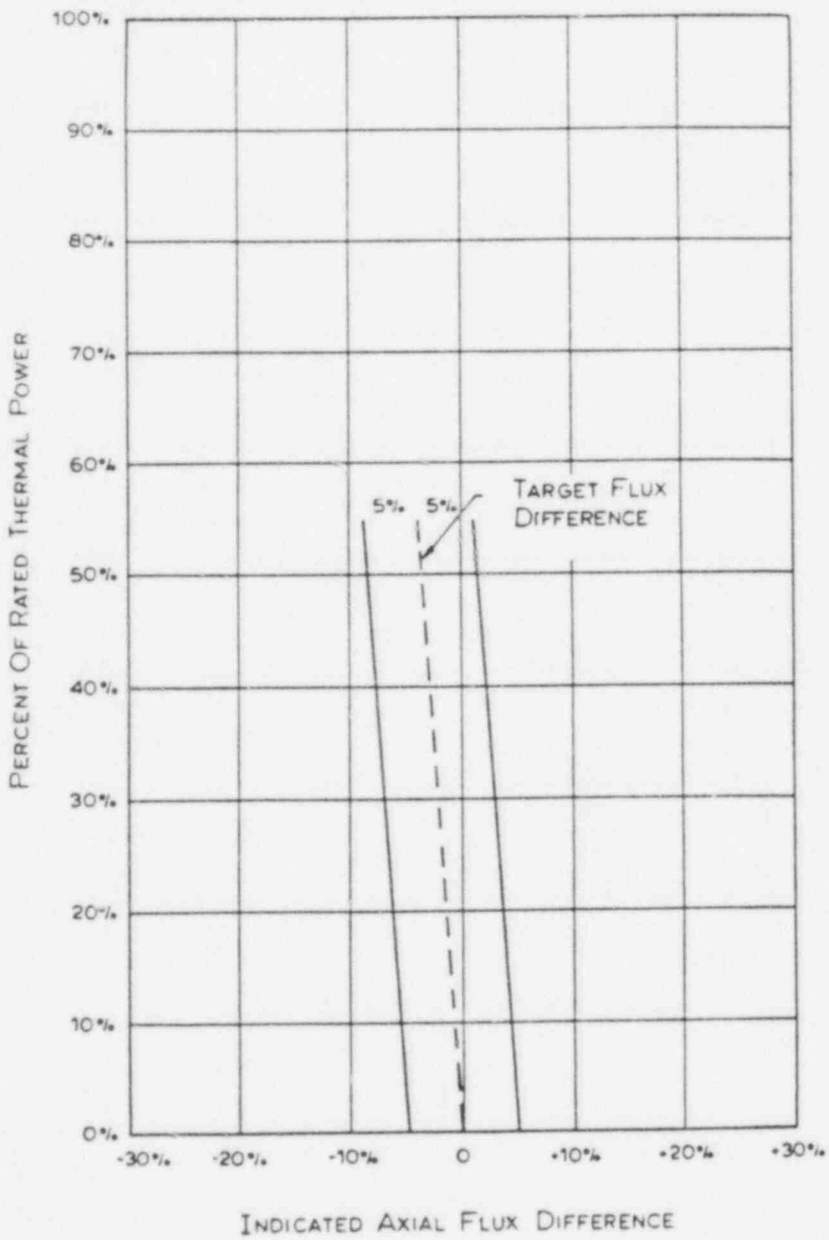


FIGURE B 3/4 2-1b
TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER
FOR THREE LOOP OPERATION

HEAT FLUX HOT CHANNEL FACTOR and RCS 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. 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.

The $F^N_{\Delta H}$ as calculated in Specifications 3.2.3.1 and 3.2.3.2 are 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.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following:

a. Design limit DNBR of 1.30 vs 1.28,

b. Grid Spacing (K_s) of 0.046 vs 0.059,

c. Thermal Diffusion Coefficient of 0.038 vs 0.059,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of the Limiting Condition for Operation. Measurement errors of 2.4% for four loop flow and 2.76% for three loop flow for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS 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 non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi will be added if venturis are not verified clean every 18 months. Any fouling which might bias the RCS 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 RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation defined in Specifications 3.2.3.1 and 3.2.3.2.

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 limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

QUADRANT POWER TILT RATIO (Continued)

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 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 flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

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 minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of 590.7°F (four loops operating) or 583.7°F (three loops operating) and the indicated pressurizer pressure value of 2208 psia (four loops or three loops operating) 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 have been accounted for in determining the parameter limits.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION 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 is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) 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.

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 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

REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

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. Detector response times may be measured by the in situ on line noise analysis-response time degradation method described in the Westinghouse Topical Report, "The Use of Process Noise Measurements To Determine Response Characteristics of Protection Sensors in U.S. Plants," August 1983.

The Engineered Safety Features Acutation 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) quench spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start, (10) service water pumps start and automatic valves position, and (11) Control Room isolates.

REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

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 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure and low steam line pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure and low steam line pressure.

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves, main feed pumps and main 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 thi, 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 INSTRUMENTATION

The OPERABILITY of the Remote Shutdown Instrumentation 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 Instrumentation ensures that a fire will not preclude achieving safe shutdown. The remote shutdown monitoring

REMOTE SHUTDOWN INSTRUMENTATION (Continued)

instrumentation, 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 shut down 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. The instrumentation included in this specification are those instruments provided to monitor key variables, designated as Category 1 instruments following the guidance for classification contained in Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

The OPERABILITY of the fire 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 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.

3/4.3.3.8 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the Loose-Part Detection System 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.

3/4.3.3.9 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 REMODCM 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. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

3/4.3.3.10 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 REMODCM 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. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-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 in MODES 1 and 2 with three or four reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. With less than the required reactor coolant loops 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 core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR 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 RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR 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 an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS 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 RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) 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 RCS cold leg temperatures.

The requirement to maintain the isolated loop stop valves shut with power removed ensures that no reactivity addition to the core could occur due to the startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. Draining and refilling the isolated loop within 4 hours prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. 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 RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Cold Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. 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 RCS 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 RCS 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. Requiring the PORVs to be OPERABLE ensures that the capability for depressurization during safety grade cold shutdown is met.

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 RCS will be maintained. The program for inservice inspection of steam 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 promptly reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days and 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 RCS 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 RCS, 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 RCS 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 RCS pressure of 2250 psia. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

OPERATIONAL LEAKAGE (Continued)

The specified allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series 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, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS 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 ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure 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

SPECIFIC ACTIVITY (Continued)

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 Millstone 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 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.

SPECIFIC ACTIVITY (Continued)

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. 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. Also, the 10 CFR 50, Appendix G rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. This rule states the minimum metal temperature of the closure flange regions should be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20% of the preservice hydrostatic test pressure (636 psia). The minimum temperature of the closure flange and vessel flange regions is 150°F since the limiting RT_{NDT} is 30°F (See Table B 3/4.4-1). The heatup curve shown in Figure 3.4-2 is not impacted by the 10 CFR 50 rule. However, the cooldown curve shown in Figure 3.4-3 is impacted by the rule.

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

PRESSURE/TEMPERATURE LIMITS (Continued)

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. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice 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 testing of the ferritic materials in the reactor vessel were performed in accordance with the 1973 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 10 effective full power years (EFPY) of service life. The 10 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 10 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

TABLE B 3/4.4-1

REACTOR VESSEL FRACTURE TOUGHNESS PROPERTIES

Component	Code No.	Grade	Cu (%)	P (%)	T (°F)	50 ft-lb 35 Mil (°F)	RT (°F)	Avg. Upper Shelf NMWD (ft-lb)	Avg. Upper Shelf MWD (ft-lb)
Closure Head Dome	B9812-1	A533B, CL.1	0.08	0.010	-40	60	0	96.0	---
Closure Head Torus	B9813-1	A533B, CL.1	0.11	0.010	-40	70	10	107.5	---
Closure Head Flange	B9803-1	A508, CL.2	---	0.010	30	<90	30	121.0	---
Vessel Flange	B9801-1	A508, CL.2	0.15	0.007	-40	<20	-40	116.5	---
Inlet Nozzle	B9806-3	A508, CL.2	0.09	0.009	10	<70	10	162.0	---
Inlet Nozzle	B9806-4	A508, CL.2	0.09	0.009	0	<60	0	158.0	---
Inlet Nozzle	R5-3	A508, CL.2	0.07	0.008	-10	<50	-10	130.0	---
Inlet Nozzle	R5-4	A508, CL.2	0.08	0.009	0	<60	0	136.0	---
Outlet Nozzle	R6-1	A508, CL.2	---	0.012	-40	<20	-40	128.0	---
Outlet Nozzle	R6-2	A508, CL.2	---	0.006	-30	<30	-30	127.0	---
Outlet Nozzle	B9807-1	A508, CL.2	---	0.006	-30	<30	-30	121.0	---
Outlet Nozzle	B9807-2	A508, CL.2	---	0.005	-30	<30	-30	126.0	---
Nozzle Shell	B9804-1	A533B, CL.1	0.05	0.012	-40	100	40	85.5	---
Nozzle Shell	B9804-2	A533B, CL.1	0.08	0.010	-40	80	20	104.5	---
Nozzle Shell	B9804-3	A533B, CL.1	0.05	0.009	-50	60	0	103.5	---
Inter. Shell	B9805-1	A533B, CL.1	0.05	0.010	-40	120	60	92.5	113.5
Inter. Shell	B9805-2	A533B, CL.1	0.05	0.014	-60	70	10	90.0	129.0
Inter. Shell	B9805-3	A533B, CL.1	0.04	0.009	-40	60	0	106.5	136.5
Lower Shell	B9820-1	A533B, CL.1	0.07	0.006	-50	70	10	76.5	124.5
Lower Shell	B9820-2	A533B, CL.1	0.06	0.008	-30	100	40	75.5	114.5
Lower Shell	B9820-3	A533B, CL.1	0.05	0.007	-30	80	20	79.5	124.0
Bottom Head Torus	B9816-1	A533B, CL.1	0.13	0.012	-50	20	-40	91.5	---
Bottom Head Dome	B9817-1	A533B, CL.1	0.15	0.012	-30	<30	-30	161.0	---
Inter & Lower Shell	G1.59	SAW	0.07	0.011	-50	---	-50	200	---
Long & Girth Weld Seams	---	---	---	---	---	---	---	---	---

NOTES:
NMWD = normal to major working direction
MWD = major working direction

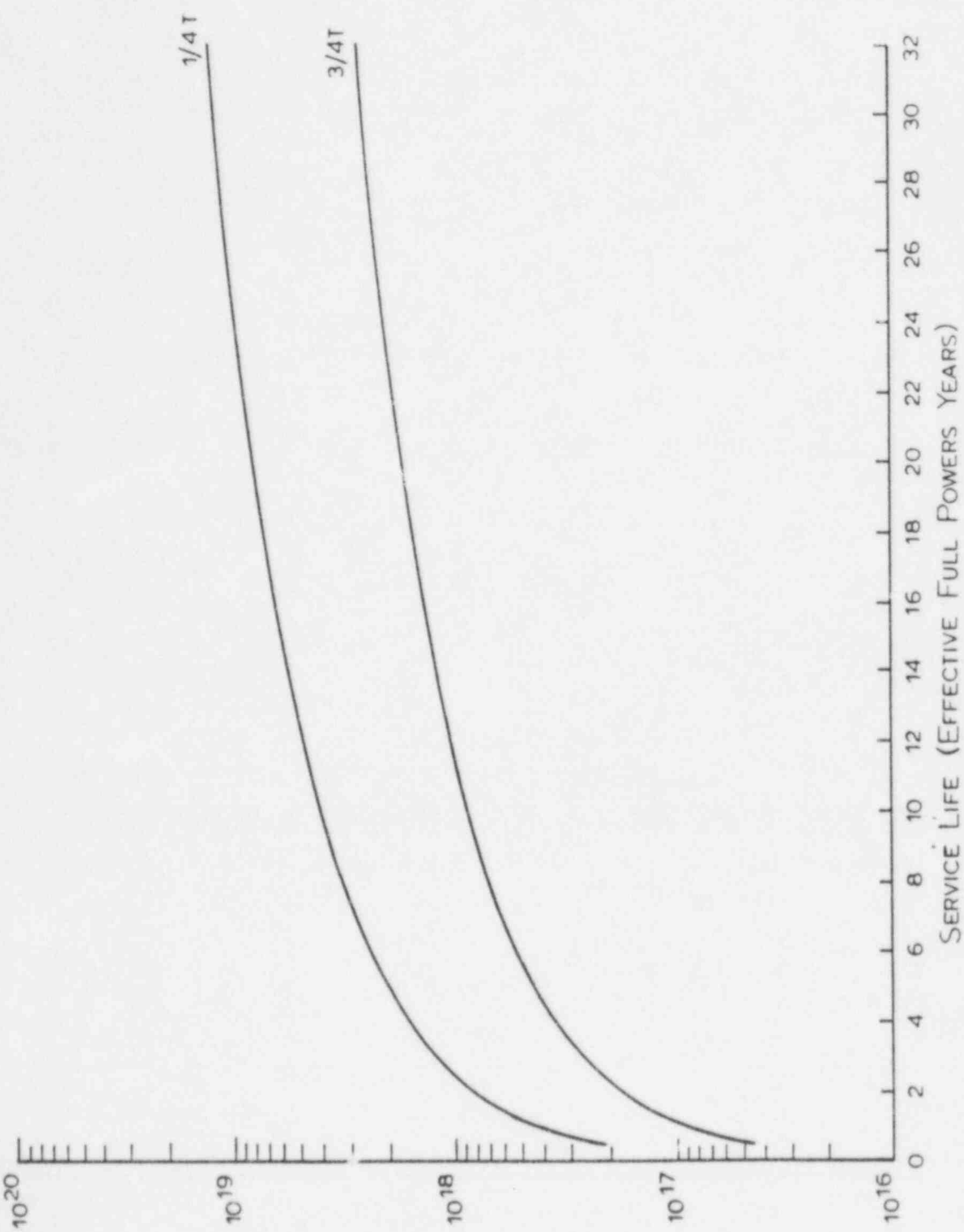


FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

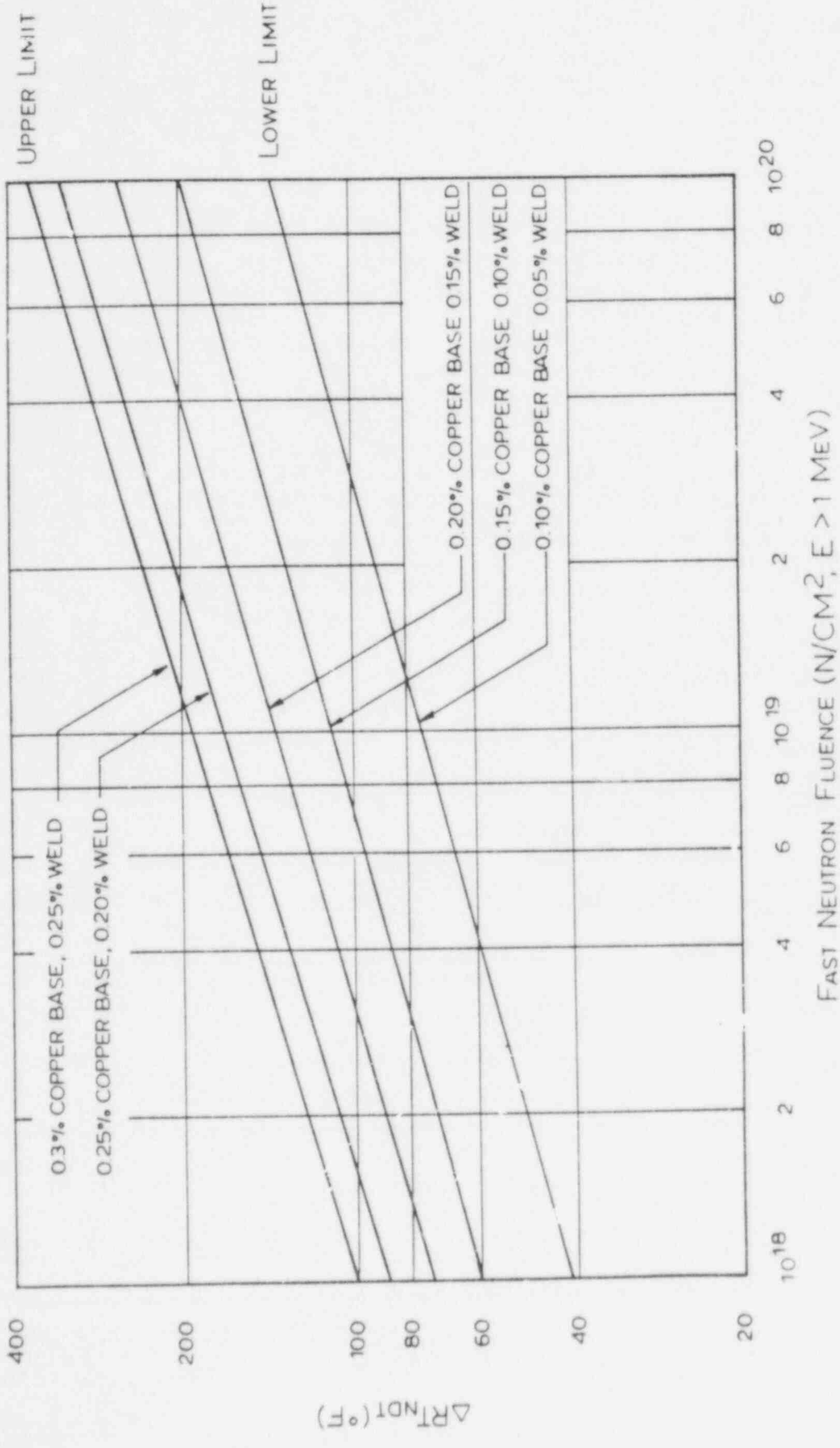


FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSELS EXPOSED TO 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

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 reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor 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.

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:

PRESSURE/TEMPERATURE LIMITS (Continued)

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

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:

$$C\ K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

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.

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

PRESSURE/TEMPERATURE LIMITS (Continued)

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 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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.

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.

COLD OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 7.0 square inches ensures that the RCS 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 RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a charging pump and its injection into a water-solid RCS.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Protection System (COPS) is derived by analysis which models the performance of the COPS 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 an RCP if secondary temperature is more than 50°F above primary temperature.

The Maximum Allowed PORV Setpoint for the COPS 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, 80 Edition and Addenda through Winter except where specific written relief has been granted 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 Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that 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 vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS 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 RCS 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 RCS 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 charging pump to be inoperable below 350°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 (RWST) 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 RWST 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 RWST and the RCS 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 RWST also ensure a pH value of between 7.0 and 7.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The maximum/minimum solution temperatures for the RWST in MODES 1, 2, 3 and 4 are based on analysis assumptions.

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 guidelines 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 during performance of the periodic test 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 provides 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 and 3/4.6.1.5 AIR PARTIAL PRESSURE and AIR TEMPERATURE

The limitations on containment air partial pressure and average air temperature as a function of service water temperature ensure that: (1) the containment structure is prevented from exceeding its design negative pressure of 8 psia, (2) the containment peak pressure does not exceed the design pressure of 60 psia during LOCA conditions, and (3) the containment pressure is returned to subatmospheric conditions following a LOCA within 60 minutes. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

The limits on the parameters of Figure 3.6-1 are consistent with the assumptions of the safety analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 60 psia in the event of a LOCA. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be locked closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves 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 locked 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.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply 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

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH SPRAY SYSTEM and RECIRCULATION SPRAY SYSTEM

The OPERABILITY of the Containment Spray Systems ensures that containment depressurization and subsequent return to subatmospheric pressure will occur in the event of a LOCA. The pressure reduction and resultant termination of containment leakage are consistent with the assumptions used in the safety analyses.

3/4.6.2.3 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 7.0 and 7.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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 General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for these 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 or the Mechanical Vacuum Pumps are 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 in Containment Following a LOCA," March 1971.

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

3/4.6.5.1 STEAM JET AIR EJECTOR

The closure of the isolation valves in the suction of the steam jet air ejector ensures that: (1) the containment internal pressure may be maintained within its operation limits by the mechanical vacuum pumps, and (2) the containment atmosphere is isolated from the outside environment in the event of a LOCA. These valves are required to be closed for containment isolation.

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

The OPERABILITY of the Supplementary Leak Collection and Release System ensures that containment leakage occurring during LOCA conditions into the enclosure building will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. Cumulative operation of the system with the heaters operating 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.

3/4.6.6.2 ENCLOSURE BUILDING INTEGRITY

Secondary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the Supplementary Leak Collection and Release System, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.6.3 ENCLOSURE BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment enclosure building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide 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.

3/4.7.1 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% (1305 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 100% RATED THERMAL POWER 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 1.579×10^7 lbs/h which is 105% of the total secondary steam flow of 1.504×10^7 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 3.7-2.

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

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

For three loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times 80$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

U = Maximum number of inoperable safety valves per operating steam line, SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

80 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for three 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 or accident conditions coincident with a total loss-of-offsite power.

The auxiliary feedwater system is capable of delivering a total feedwater flow of 480 gpm at a pressure of 1236 psia to the entrance of at least three steam generators while allowing for (1) any spillage through the design worst-case break of the Normal feedwater line, (2) the design worst-case single failure; and (3) recirculation flow. 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 at which point the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK

The OPERABILITY of the demineralized water storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 10 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power, and with an additional 6-hour cooldown period to reduce reactor coolant temperature to 350°F. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 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 primary-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.

3/4.7.1.5 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.

3/4.7.2 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 REACTOR PLANT COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Reactor Plant 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 SERVICE WATER SYSTEM

The OPERABILITY of the 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 analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitation on the ultimate heat sink temperature ensures that cooling water at less than the design temperature limit is available to either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

ULTIMATE HEAT SINK (Continued)

The limitation on maximum temperature is 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 FLOOD PROTECTION

The limitation on flood protection ensures that the service water pump cubicle watertight doors will be closed before the water level reaches the critical elevation of 14.5 feet Mean Sea Level. Elevation 14.5 feet MSL is the level at which external flood waters could enter the service water pump cubicle.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency 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 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 for the duration of the accident. 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.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM

The OPERABILITY of the two independent Control Room Envelope Pressurization Systems ensures that: (1) breathable air is supplied to the control room, instrumentation rack room, and computer room, and (2) a positive pressure is maintained within the control room envelope during control building isolation. Each system will provide air to the control room for 1 hour following an initiation of a control building isolation signal at which time, the Control Room Emergency Ventilation System would be started.

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

The OPERABILITY of the Auxiliary Building Filter System ensures that radioactive materials leaking from the equipment within the charging pump,

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM (Continued)

component cooling water pump and heat exchanger areas following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating 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.

3/4.7.10 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. For the purpose of declaring the affected system OPERABLE with the inoperable snubber(s), an engineering evaluation may be performed, in accordance with Section 50.59 of 10 CFR Part 50.

Snubbers are classified and grouped by design and manufacturer but not by size. Snubbers of the same manufacturer but having different internal mechanisms are classified as different types. 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 Plant Operations Review Committee. 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 deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is 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

SNUBBERS (Continued)

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. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability, one of three functional testing methods is used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 5% 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 destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

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.

3/4.7.11 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

3/4.7.11 SEALED SOURCE CONTAMINATION (Continued)

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.

3/4.7.12 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, Halon, fire hose stations, and yard fire hydrants.

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.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method.

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.

3/4.7.13 FIRE RATED ASSEMBLIES

The functional integrity of the fire rated assemblies and 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

FIRE RATED ASSEMBLIES (Continued)

a passive element in the facility Fire Protection Program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers 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.

3/4.7.14 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ± 2.2°F.

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 Guides 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; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement 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 4.8-2a 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 4.8-2a 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 provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The OPERABILITY of the motor-operated valves thermal overload protection and integral bypass devices ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of the thermal overload protection are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

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 operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS 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 ensure that a release of radioactive material within containment will be restricted from leakage to the environment. 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.

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 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) refueling machines will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive 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 AREAS

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 (RHR) 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 RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR 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 pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 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.

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

The limitations on the Fuel Building Exhaust Filter System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating 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 RCS 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 which in turn may cause the RCS T_{avg} to 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 from the site will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to an individual and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for 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.

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 will be kept "as low as is reasonably achievable." The dose calculation methodology and parameters in the REMODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the REMODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be 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 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

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 from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for all areas offsite. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual offsite to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR 20.106(b)). For individuals who may at times be within the SITE BOUNDARY, the occupancy of that individual will usually be

DOSE RATE (Continued)

sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual 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 cow-milk-child pathway to less than or equal to 1500 mrems/year.

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 will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The dose calculation established in the REMODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be 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 REMODCM equations provided for determining the air doses at the site boundary are based on utilizing successively more realistic dose calculational methodologies. More realistic dose calculational methods are used whenever simplified calculations indicate a dose approaching a substantial portion of the regulatory limits. The methods used, in order, are previously determined air dose per released activity ratio, historical meterological data and actual radionuclide mix released, or real time meteorological and actual radionuclides released.

3/4.11.2.3 DOSE - RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

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

DOSE - RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES (Continued)

that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The REMODCM 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 an individual through appropriate pathways is unlikely to be substantially underestimated. The REMODCM 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. The release rate specifications for radioiodines and radionuclides in particulate form and radionuclides other than noble gases are dependent upon the existing radionuclide pathways to man. The pathways that are examined in the development of these calculations are: (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 to man.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190. For the purposes of the Special Report, it may be assumed that the dose commitment to any REAL 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 5 miles must be considered.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS

5.1.3 The site boundary for liquid and gaseous effluents shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment building is a steel-lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 201 feet, 3 inches.

c. Minimum thickness of concrete walls = 4 feet, 6 inches.

d. Minimum thickness of concrete roof = 2 feet, 6 inches.

e. Minimum thickness of concrete floor pad = 10 feet.

f. Nominal thickness of steel liner = 1/4 inch (floor), 3/8 inch (wall), and 1/2 inch (dome).

g. Net free volume = 2.26×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a minimum internal pressure of 8 psia, a maximum internal pressure of 59.7 psia, and a temperature of 280°F.

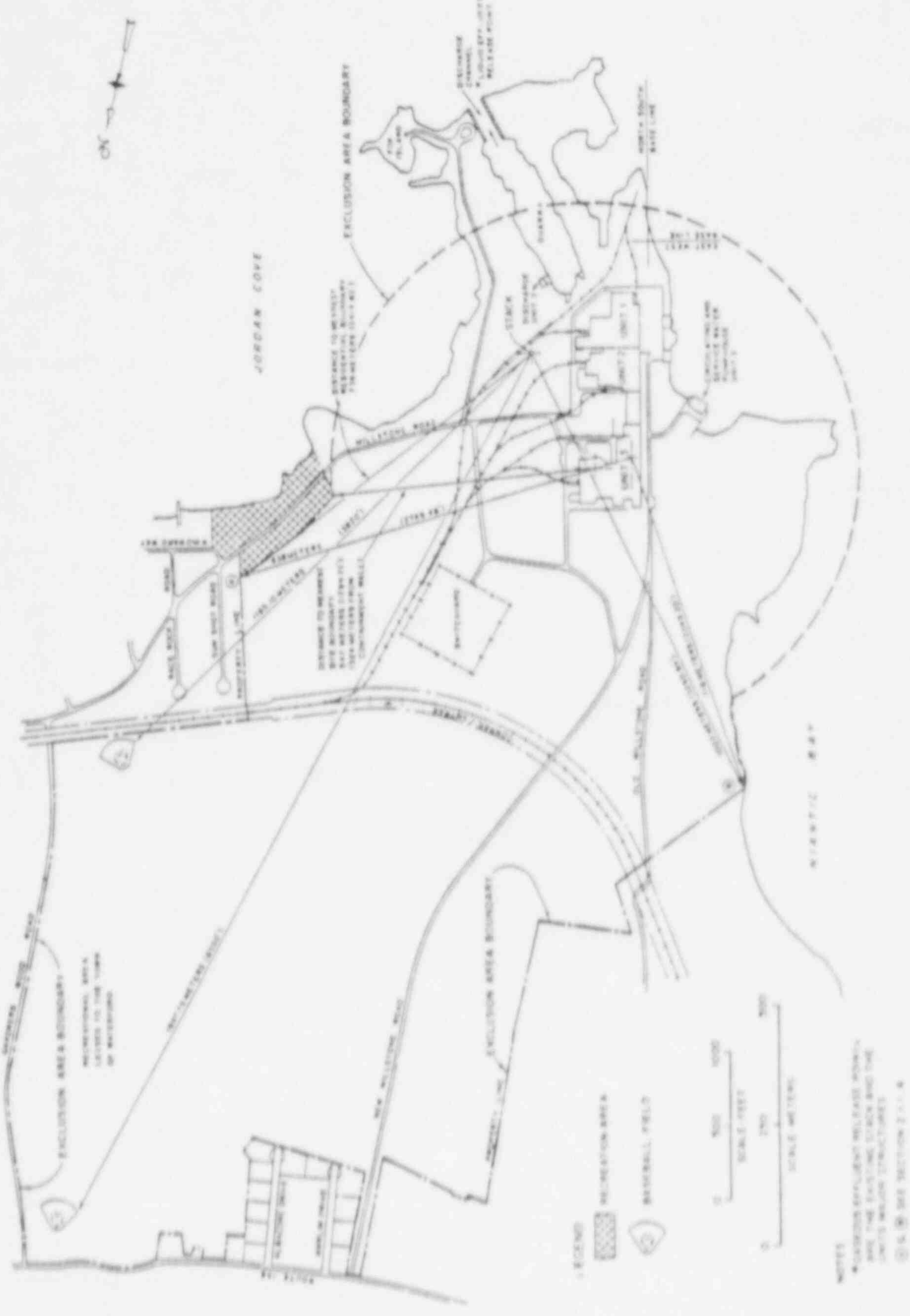


FIGURE 5.1-1 EXCLUSION AREA

FIGURE 5.1-2 LOW POPULATION ZONE

FIGURE 5.1-3 SITE BOUNDARY FOR LIQUID AND GASEOUS EFFLUENTS

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 1810.7 grams uranium. The initial core loading shall have a maximum enrichment of 3.4 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.8 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 61 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 95.3% hafnium and 4.5% natural zirconium. All control rods shall be clad with stainless steel.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, 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 12,240 cubic feet at a nominal T_{avg} of 587°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-3.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The 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 of 2.6% $\Delta k/k$ for uncertainties as described in Section 4.3 of the FSAR, and

b. A nominal 10.35-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 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 756 PWR 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.
	200 leak tests.	Pressurized to $\geq$ 2500 psia.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 3125 psia.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 1350 psig.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Station Superintendent shall be responsible for overall operation of the Millstone Station site while the Unit Superintendent shall be responsible for operation of the unit. The Station Superintendent and Unit Superintendent shall each delegate in writing the succession to these responsibilities during their absence.

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 shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. A Health Physics Technician* shall be on site when fuel is in the reactor;

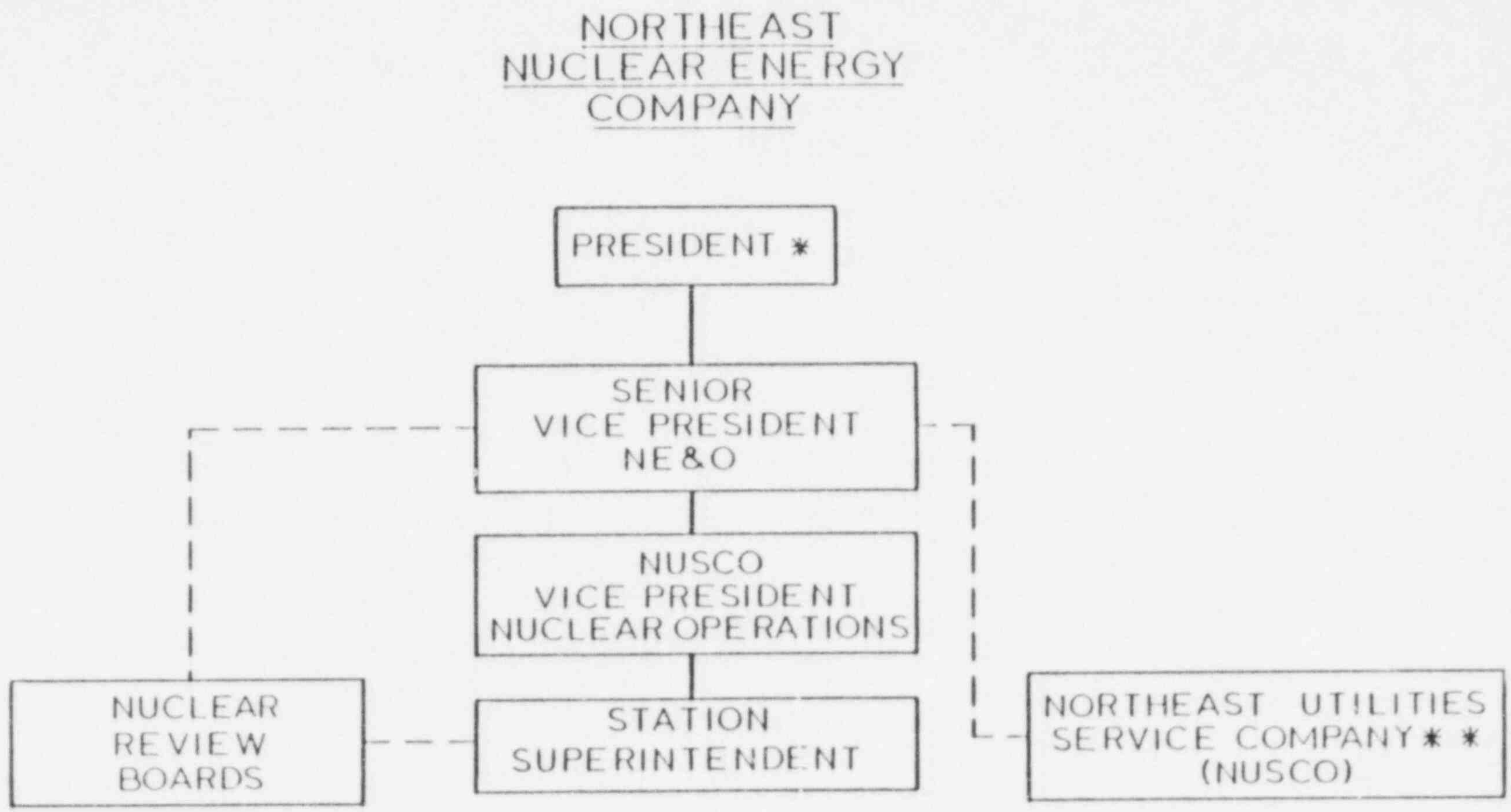
e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

f. A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include two 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)

g. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions. These procedures should follow the general guidance of the NRC Policy Statement on working hours (Generic Letter No. 82-12).



* OVERALL CORPORATE RESPONSIBILITY FOR FIRE PROTECTION

* * PROVIDES OPERATING AND ENGINEERING SUPPORT BY CONTRACTUAL ARRANGEMENT

FIGURE 6.2-1 OFFSITE ORGANIZATION

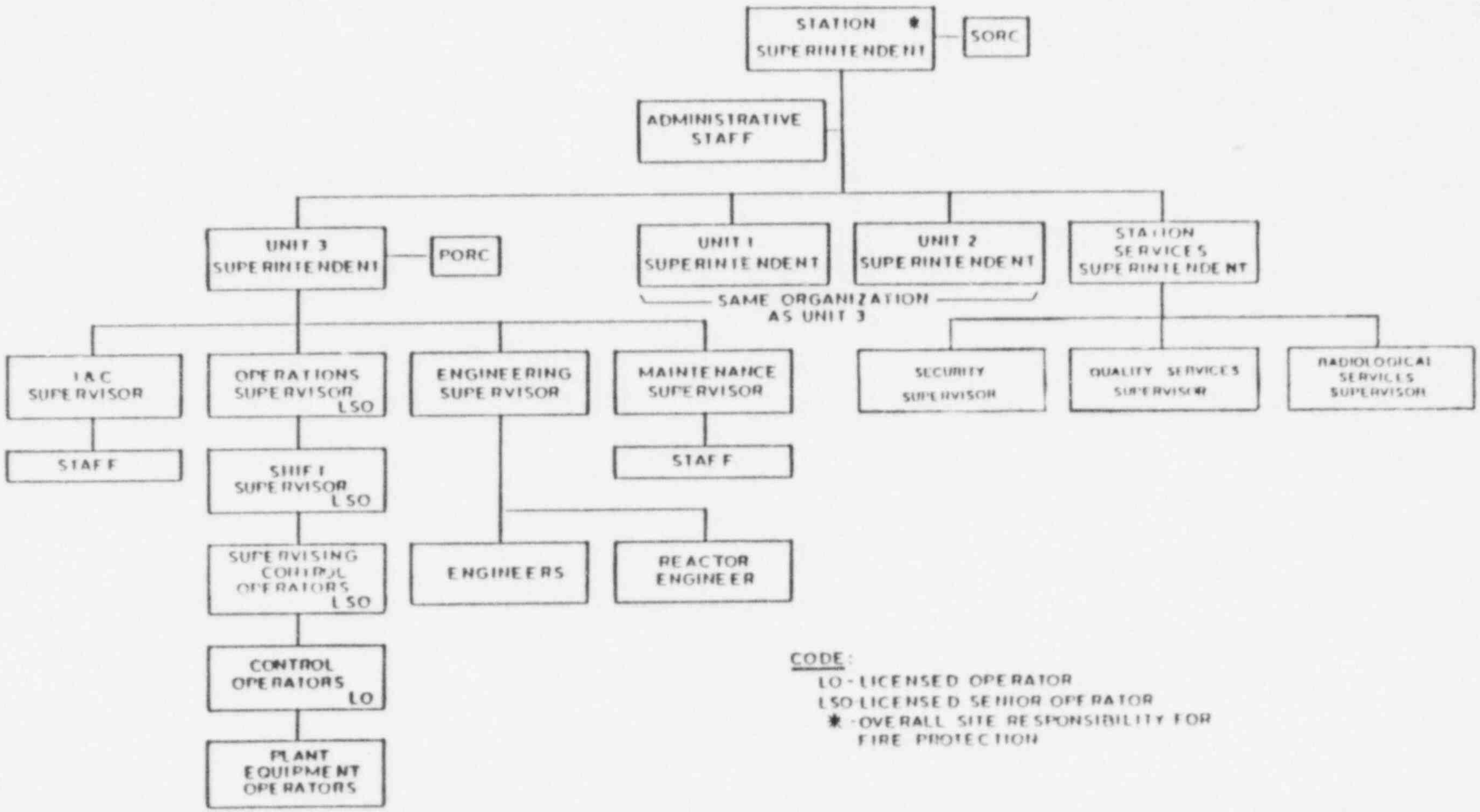


FIGURE 6.2-2 UNIT ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1	1
SRO	1	None
RO	2	1
PEO	2	1
STA	1*	None

SS - Shift Supervisor with a Senior Operator license on Unit 3
SRO - Individual with a Senior Operator license on Unit 3
RO - Individual with an Operator license on Unit 3
PEO - Plant Equipment Operator (Non-licensed)
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 snift 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 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.

*The STA position may be filled by an on-shift Senior Reactor Operator only if that Senior Reactor Operator meets the Shift Technical Advisor qualifications of the Commission Policy Statement on Engineering Expertise on Shift.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall include, as part of its function, examination of unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Vice President-Nuclear and Environmental Engineering.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least four full-time personnel located on site to perform the functions described in 6.2.3.1 for Millstone Unit 3. Each person shall have either:

(1) A bachelor's degree in engineering or related science and at least 2 years of professional level experience in his field, at least 1 year of which experience shall be in the nuclear field, or,

(2) At least 10 years of professional level experience in his field, at least 5 years of which experience shall be in the nuclear field.

A minimum of 50% of these personnel shall have the qualifications specified in (1) above.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit 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 ISEG shall be prepared and maintained, and quarterly reports of completed safety evaluations will be made to the Vice President-Nuclear and Environmental Engineering.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

---

*Not responsible for sign-off function.

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, Revision 1, May 1977. 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 Superintendent 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.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Unit Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Unit Superintendent
Vice Chairman and Member:	Operations Supervisor
Member:	Maintenance Supervisor
Member:	Instrument and Control Supervisor
Member:	Reactor Engineer
Member:	Engineering Supervisor or Startup Supervisor*
Member:	Station Services Superintendent or Quality Services Supervisor or Radiological Services Supervisor
Member:	Staff Engineer**

*When position is staffed.

**The Staff Engineer member of the PORC shall have an academic degree in engineering or physical science field; and, in addition, shall have a five years technical experience, of which a minimum of three years shall be in the nuclear power plant industry.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman.

QUORUM

6.5.1.5 The quorum of the PORC shall consist of the Chairman or Vice Chairman or Station Superintendent and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of: (1) all procedures, except common site procedures, required by Specification 6.8 and changes thereto, and (2) any other proposed procedures or changes thereto as determined by the Unit Superintendent to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Sections 1.0-5.0 of these Technical Specifications;

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Vice President-Nuclear Operations and to the Chairman of the Nuclear Review Board;

f. Review of all REPORTABLE EVENTS;

g. Review of facility operations to detect potential safety hazards;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Chairman of the Nuclear Review Board or the Station Superintendent; and

i. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through d. above constitutes an unreviewed safety question.

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend to the Unit Superintendent written approval or disapproval of items considered under Specification 6.5.1.6a. through d. above; and

b. Provide written notification to the Station Superintendent, Vice President-Nuclear Operations and the Chairman of the Nuclear Review Board of disagreement between the PORC and the Unit Superintendent; however, the Unit Superintendent shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each meeting and copies shall be provided to the Station Superintendent, Vice President-Nuclear Operations and the Chairman of the Nuclear Review Board.

6.5.2 SITE OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.2.1 The SORC shall function to advise the Station Superintendent on all matters related to nuclear safety of the entire Millstone Station Site.

COMPOSITION

6.5.2.2 The SORC shall be composed of the:

Chairman:	Station Superintendent
Member:	Unit 1 Superintendent
Member:	Unit 2 Superintendent
Member:	Unit 3 Superintendent
Member:	Designated Member of Unit 1 PORC
Member:	Designated Member of Unit 2 PORC
Member:	Designated Member of Unit 3 PORC
Member:	Station Services Superintendent

ALTERNATES

6.5.2.3 Alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SORC activities at one time.

MEETING FREQUENCY

6.5.2.4 The SORC shall meet at least once per 6 months and as convened by the SORC Chairman.

QUORUM

6.5.2.5 A quorum of the SORC shall consist of the Chairman and four members including alternates.

RESPONSIBILITIES

6.5.2.6 The SORC shall be responsible for:

a. Review of (1) all common site procedures required by Specification 6.8 and changes thereto, (2) any other proposed procedures or changes thereto as determined by the Station Superintendent to affect site nuclear safety;

b. Review of all proposed changes to Section 6.0 "Administrative Controls" of these Technical Specifications;

c. Performance of special reviews and investigations and reports as requested by the Chairman of the Site Nuclear Review Board;

d. Review of the Plant Security Plan and implementing procedures and submittal of recommended changes to the Chairman of the Site Nuclear Review Board;

e. Review of the Emergency Plan and implementing procedures, and submittal of recommended changes to the Chairman of the Site Nuclear Review Board;

f. Review of all common site proposed tests and experiments that affect nuclear safety;

g. Review of all common site proposed changes or modifications to systems or equipment that affect nuclear safety; and

h. Render determinations in writing or meeting minutes with regard to whether or not each item considered under Specification 6.5.2.6(a) through (g) above constitutes an unreviewed safety question.

AUTHORITY

6.5.2.7 The SORC shall:

a. Recommend to the Station Superintendent written approval or disapproval in meeting minutes of items considered under Specification 6.5.2.6(a) through (g) above, and

b. Provide immediate written notification or meeting minutes to the Vice President-Nuclear Operations and the Chairman of the Site Nuclear Review Board of disagreement between the SORC and the Station Superintendent; however, the Station Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.2.8 The SORC shall maintain written minutes of each meeting and copies shall be provided to the Vice President-Nuclear Operations and Chairman of the Site Nuclear Review Board.

6.5.3 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.3.1 The NRB 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,

d. Metallurgy,

e. Nondestructive testing,

f. Instrumentation and control,

g. Radiological safety, and

h. Mechanical and electrical engineering.

COMPOSITION

6.5.3.2 The NRB shall consist of no less than the seven, nor more than eleven members including the Chairman and the Unit Superintendent. The Chairman and members of the NRB shall be appointed in writing by the Senior Vice President-Nuclear Engineering and Operations and shall have an academic degree in an engineering or physical science field or hold a senior management position. In addition, they shall have a minimum of five years technical experience, of which a minimum of three years shall be in their respective field of expertise.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NRB Chairman to provide expert advice to the NRB.

MEETING FREQUENCY

6.5.3.4 The NRB shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.3.5 The quorum of the NRB necessary for the performance of the NRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least enough members to constitute a majority of the assigned members. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.3.6 The NRB shall review:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provisions of 10 CFR 50.59, 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 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes in Sections 1.0-5.0 of these 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 plant equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. Indications of a significant unanticipated deficiency, affecting nuclear safety, in some aspect of design or operation of structures, systems, or components; and

i. Reports and meeting minutes of the PORC.

AUDITS

6.5.3.7 Audits of unit activities shall be performed under the cognizance of the NRB. These audits shall encompass:

a. The conformance of unit operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training, and qualifications of the unit staff at least once per 12 months;

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. Any other area of unit operation considered appropriate by the NRB or the Senior Vice President-Nuclear Engineering and Operations.

AUTHORITY

6.5.3.8 The NRB shall report to and advise the Senior Vice President Nuclear Engineering and Operations on those areas of responsibility specified in Sections 6.5.3.6 and 6.5.3.7. Meeting minutes may be used for this purpose.

RECORDS

6.5.3.9 Records of NRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NRB meeting shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Engineering and Operations within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.3.6 shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Engineering and Operations within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.3.7 shall be forwarded to the Senior Vice President-Nuclear Engineering and Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.5.4 SITE NUCLEAR REVIEW BOARD (SNRB)

FUNCTION

6.5.4.1 The SNRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations

b. Administration

c. Chemistry and radiochemistry

d. Quality Assurance practices

e. Radiological safety

COMPOSITION

6.5.4.2 The SNRB shall consist of no less than six, nor more than nine members including the individual Millstone NRB Chairman and the Station Superintendent. The Chairman and members of the SNRB shall be appointed in writing by the Senior Vice President-Nuclear Engineering and Operations, and shall have an academic degree in engineering or physical science field or hold a senior management position. In addition, they shall have a minimum of five years technical experience, of which a minimum of three years shall be in their respective field of expertise.

CONSULTANTS

6.5.4.3 Consultants shall be utilized as determined by the SNRB Chairman to provide expert advice to the SNRB.

MEETING FREQUENCY

6.5.4.4 The SNRB shall meet at least once per calendar year.

QUORUM

6.5.4.5 A quorum of SNRB shall consist of the Chairman or his designated alternate and four SNRB members. No more than a minority of the quorum shall have line responsibility for operation of the Station.

REVIEW

6.5.4.6 The SNRB shall review:

a. Proposed changes in Section 6.0 of these Technical Specifications or Licenses common to all units.

b. Any indication of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems or components common to all units.

c. Reports and meeting minutes of the SORC.

AUDITS

6.5.4.7 Audits of site activities shall be performed under the cognizance of the SNRB. These audits shall encompass:

a. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

b. The Site Emergency Plan and implementing procedures at least once per 24 months.*

c. The Site Security Plan and implementing procedures at least once per 24 months.*

d. The Fire Protection Program and implementing procedures at least once per 24 months.

e. An inspection and audit of the fire protection and loss prevention program shall be performed at least once per 12 months by an outside firm experienced in fire protection and loss prevention.

f. The performance of activities in accordance with the RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL at least once per 24 months;

g. The performance of activities required by the quality control section of Regulatory Guide 1.21, Rev. 1, June 1974, and Regulatory Guide 4.1, Rev. 1, April 1975, at least once per 12 months.

---

*Annual audits of the Emergency Plan and the Security Plan are also performed to satisfy the additional requirements of 10 CFR Part 50.54(t) and 10 CFR Part 7.40(d), respectively. These annual audits may be used to satisfy the requirements contained in the Technical Specifications.

AUTHORITY

6.5.4.8 The SNRB reports to and advises the Senior Vice President-Nuclear Engineering and Operations on those areas of responsibility specified in Sections 6.5.4.6 and 6.5.4.7. Meeting minutes may be used for this purpose.

RECORDS

6.5.4.9 Records of SNRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each SNRB meeting shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Engineering and Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.4.6 above shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Engineering and Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.4.7 above shall be forwarded to the Senior Vice President-Nuclear Engineering and Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

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 PORC, and the results of this review shall be submitted to the Chairman of the NRB and the Vice President-Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The Unit shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear Operations and the Chairman of the NRB shall be notified within 24 hours;

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. 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;

SAFETY LIMIT VIOLATION (Continued)

d. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the NRB, and the Vice President-Nuclear Operations within 14 days of the violation; and

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 applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto;

c. Refueling operations;

d. Surveillance activities of safety related equipment;

e. Security Plan implementation;

f. Emergency Plan implementation;

g. Fire Protection Program implementation;

h. Quality controls for effluent monitoring, using the guidance in Regulatory Guide 1.21, Rev. 1, June 1974; and

i. Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) implementation except for Section I.E, Radiological Environmental Monitoring.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the PORC/SORC, as appropriate, and shall be approved by the Unit Superintendent/Station Superintendent prior to implementation and 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

PROCEDURES AND PROGRAMS (Continued)

c. The change is documented, reviewed by the PORC/SORC, as appropriate, and approved by the Unit Superintendent/Station Superintendent 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 recirculation spray, Safety Injection, charging portion of chemical and volume control, and hydrogen recombiners. 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) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, 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,

PROCEDURES AND PROGRAMS (Continued)

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. 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.

e. Accident Monitoring Instrumentation

A program which will ensure the capability to monitor plant variables and systems operating status during and following an accident. This program shall include those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials (Category 2 and 3 instrumentation as defined in Regulatory Guide 1.97, Revision 2) and provide the following:

1) Preventive maintenance and periodic surveillance of instrumentation,

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

6.8.5 Written procedures shall be established, implemented and maintained covering Section I.E, Radiological Environmental Monitoring, of the REMODCM.

6.8.6 All procedures and procedure changes required for the Radiological Environmental Monitoring Program of Specification 6.8.5 above shall be reviewed by an individual (other than the author) from the Radiological Assessment Branch or the Production Operation Services Laboratory (POSL) and approved by appropriate supervision.

Temporary changes may be made provided the intent of the original procedure is not altered and the change is documented and reviewed by an individual (other than the author) from the Radiological Assessment Branch or the POSL, within 14 days of implementation.

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 Regional Office of the NRC 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.

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.

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*

6.9.1.2 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

*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.

ANNUAL REPORTS (Continued)

year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

Reports required on an annual basis shall include:

a. 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;

b. The results of specific activity analyses in which the reactor coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioidine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.

c. Documentation of all challenges to the pressurizer power-operated relief valves (PORVs) and safety valves; and

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

6.9.1.3 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 Annual Radiological Environmental Operating Reports shall include that information delineated in the REMODCM.

---

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

**A single submittal may be made for a multiple unit station.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.4 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.

A supplemental report containing dose assessments for the previous year shall be submitted annually within 90 days after January 1.

The report shall include that information delineated in the REMODCM.

Any changes to the REMODCM shall be submitted in the Semiannual Radioactive Effluent Release Report.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience 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

6.9.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the NRC Regional Administrator with a copy to Director of 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 substantial 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.

*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.

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.

6.10.2 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.

6.10.3 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;

RECORD RETENTION (Continued)

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;

h. Records of inservice inspections performed pursuant to these Technical Specifications;

i. Records of quality assurance activities required by the Quality Assurance Topical Report not listed in Specification 6.10.2;

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 PORC, the NRB, the SORC, and the SNRB;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.10 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

6.11.1 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 Pursuant to paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c),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

HIGH RADIATION AREA (Continued)

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 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 Health Physics Supervisor 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 Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work 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 activiated as a warning device.

6.13 RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

Section I, Radiological Effluents Monitoring Manual, shall outline the sampling and analysis programs to determine the concentration of radioactive materials released offsite as well as dose commitments to individuals in those exposure pathways and for those radionuclides released as a result of station operation. It shall also specify operating guidelines for radioactive waste treatment systems and report content.

Changes to Section I shall be submitted to the Commission for approval prior to implementation.

Section II, the Offsite Dose Calculation Manual (ODCM), shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculations of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these technical specifications. Changes to Section II need not be submitted to the Commission for approval prior to implementation, but shall be included in the next Semi-Annual Radioactive Effluent Release Report.

6.14 RADIOACTIVE WASTE TREATMENT

Procedures for liquid and gaseous radioactive effluent discharges from the Unit shall be prepared, approved, maintained and adhered to for all operations involving offsite releases of radioactive effluents. These procedures shall specify the use of appropriate waste treatment systems utilizing the guidance provided in the REMODCM.

The Solid Radioactive Waste Treatment System shall be operated in accordance with the Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

*U.S. GOVERNMENT PRINTING OFFICE:1985-491-620:20329

NRC FORM 335 (6-83) U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG-1176

2. Leave blank

3. TITLE AND SUBTITLE
Technical Specifications for Millstone Nuclear Power Station, Unit No. 3

4. RECIPIENT'S ACCESSION NUMBER

5. DATE REPORT COMPLETED
MONTH: JANUARY YEAR: 1986

6. AUTHOR(S)

7. DATE REPORT ISSUED
MONTH: JANUARY YEAR: 1986

8. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

9. PROJECT/TASK/WORK UNIT NUMBER

10. FIN NUMBER

11. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 8. above

12a. TYPE OF REPORT
TECHNICAL

12b. PERIOD COVERED (Inclusive dates)
JANUARY 1986

13. SUPPLEMENTARY NOTES
Docket No. 50-423

14. ABSTRACT (200 words or less)
The Millstone Nuclear Power Station, Unit No. 3 Technical Specifications were prepared by the U. S. Nuclear Regulatory Commission to set forth limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the protection of the health and safety of the public.

15a. KEY WORDS AND DOCUMENT ANALYSIS

15b. DESCRIPTORS

16. AVAILABILITY STATEMENT
Unlimited

17. SECURITY CLASSIFICATION (This report)
UNCLASSIFIED

18. NUMBER OF PAGES

19. SECURITY CLASSIFICATION (This page)
UNCLASSIFIED

20. PRICE
$

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC
WASH. D.C.
PERMIT No. G-67