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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated July 18, 1994, as supplemented by letters dated October 9, 1995, February 13 and March 8, 1996, Duke Power Company (DPC or the licensee) proposed license amendments to change the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested change consists of splitting the combined Unit 1 and Unit 2 TS into separate volumes for each unit. The licensee had decided that splitting the combined Unit 1 and Unit 2 TS would facilitate reactor operator actions during and after steam generator (SG) replacement. Unit 1 is scheduled to replace its SGs in early 1997 and Unit 2 is scheduled for late 1997. The October 9, 1995, February 13 and March 8, 1996, letters provided clarifying information that did not change the July 18, 1994, application and initial proposed no significant hazards consideration determination.

Separating the TS into volumes for each unit will result in the following:

a. TS pages will contain the same information as before but with the exception of references to the different units on the same page (i.e., different operating parameters, setpoints, or numerical explanation for each unit). Unit 1 volume will contain parameters and setpoint values specific to Unit 1 and Unit 2 volume will contain applicable information to Unit 2.

b. TS pages identifying specific information on each unit (e.g., heatup/cooldown curves) will be found in the unit-specific volume which they are defining.

c. TS Limiting Condition for Operation Section 3.0.5 and TS Surveillance Requirement Section 4.0.6 will be deleted since each unit's specifications (as stated in a. above) will be located in a separate volume; no statements will be necessary to indicate differences in parameters between units.

These changes are considered to be administrative. The licensee also identified other administrative changes and editorial changes in addition to those brought about by the actual TS split. These changes include:

a. incorporating license amendments that have been previously approved subsequent to the July 18, 1994, application;

b. renumbering pages and deleting pages that were intentionally left blank;

c. deleting outdated footnotes; and

d. separating tables and figures specific to each unit and associated notations.

2.0 <u>EVALUATION</u>

2.1 Administrative Changes

• Specific to splitting TS into separate volumes -

These changes pertain to deleting paragraphs, sentences, words, tables, and figures in the existing TS that currently group Unit 1 and Unit 2 items together. Example:

> The Limiting Condition for Operation and Surveillance Requirements for the Pressure/Temperature limits list four figures for the heatup and cooldown curves. Figure 3.4-2 and 3.4-4 on current TS pages 3/4 4-31 and 3/4 4-33 are applicable to Unit 1 only. Figures 3.4-3 and 3.4-5 on current TS pages 3/4 4-32 and 3/4 4-34 are applicable to Unit 2 only. These figures will be renumbered as Figure 3.4-2 and Figure 3.4-3 on proposed pages 3/4 4-30 and 3/4 4-31 and located in the unit-specific volume to which they apply. The index pages will also reflect the corrected figure number. The text notations will indicate the same.

This type of change that is associated with the TS split is administrative in nature and is found to be acceptable.

• Updating the licensee's July 18, 1994, application -

The updates incorporate those amendments that have been issued since the submittal of the original application. Example:

> Subsequent to the original submittal on July 18, 1994, of the proposed TS split, Amendment Nos. 153 and 135 were issued on January 12, 1995, which revised Table 2.2-1 on page 2-4. Therefore, proposed page 2-4 was submitted by supplemental letter dated October 9, 1995, to reflect the currently issued TS for Unit 1 and Unit 2.

This type of change is administrative in nature and is found to be acceptable.

• Deleting outdated footnotes -

These footnotes were included for a one-time action during an operating cycle that has since been completed. Example:

> The existing TS page 3/4 3-45, Incore Detection System (Section 3.3.3.2) contains two footnotes that have expired. Both footnotes concern McGuire, Unit 1, Cycle 7. Since McGuire, Unit 1 is currently in Cycle 10, these two footnotes are outdated and are proposed to be removed.

This type of change is administrative in nature and is found to be acceptable.

2.2 Editorial Changes

• Renumbering of pages -

This resulted in deleting TS pages that were intentionally left blank and/or pages that had an "a," "b," or "c" designation. Example:

> Amendments 156/138 and 158/140 created intentionally blank pages when the specifications were relocated to the Selected Licensee Commitment Manual. Current pages 3/4 3-46 through 3/4 3-51, 3/4 3-78 and 3/4 3-79 are blank and considered extraneous. Since the proposed changes to the TS will result in two unit-specific volumes and require a re-issue of the entire TS, the extraneous pages are removed.

This type of change is editorial in nature and is found to be acceptable.

• Rewording -

Certain sections of the TS were revised to make wording/explanations consistent and/or clearer throughout the text. Example:

> Table 4.7-2, "Snubber Visual Inspection Interval," Note 4 on TS page 3/4 7-18b reads:
>
> > "Note 4: If the number of unacceptable snubbers is equal to or less than the Column B but greater than the number in Column A, the next inspection shall be the same as the previous interval."

For clarity and consistency to a similar note description found in Note 5 on the same page, the phrase "the Column B" will be changed to "the number in Column B." Therefore, Note 4 on proposed page 3/4 7-20 will be changed to read:

> > "Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection shall be the same as the previous interval."

This type of change is editorial in nature and is found to acceptable.

3.0 STAFF CONCLUSION

The licensee provided a summary of all proposed changes to the TS and associated Bases in their submittals. The summary included editorial and administrative changes, and changes due to TS amendments issued since the original amendment request dated July 18, 1994. For the reasons stated above, the staff finds the TS changes described in the licensee's submittals, acceptable. The corresponding separation of the TS into individual volumes for Unit 1 and Unit 2 is therefore also acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on February 14, 1996 (61 FR 5808).

Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of the amendments will not have a significant effect on the quality or the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: March 21, 1996

TECHNICAL SPECIFICATIONS

FOR

McGUIRE NUCLEAR STATION

UNIT NO. 2

DOCKET NO. 50-370

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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 with 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 operator action during periods when containment isolation valves may be opened under administrative controls pursuant to Specification 4.6.1.1.a; or

   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions.

b. All equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of 1-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

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

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

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

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

MASTER RELAY TEST

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

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by FSAR Chapter 16.

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 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.

PROCESS CONTROL PROGRAM (PCP)

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

PURGE - PURGING

1.24 PURGE or PURGING 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 required to purify the confinement.

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

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

REACTOR BUILDING INTEGRITY

1.27 REACTOR BUILDING INTEGRITY shall exist when:

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

b. The Annulus Ventilation System is in compliance with the requirements of Specification 3.6.1.8, and

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

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 to 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.

SOLIDIFICATION

1.33 Not Used

SOURCE CHECK

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

STAGGERED TEST BASIS

1.35 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

WASTE GAS HOLDUP SYSTEM

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

TABLE 1.1

FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL MODES

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

*Excluding decay heat.

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

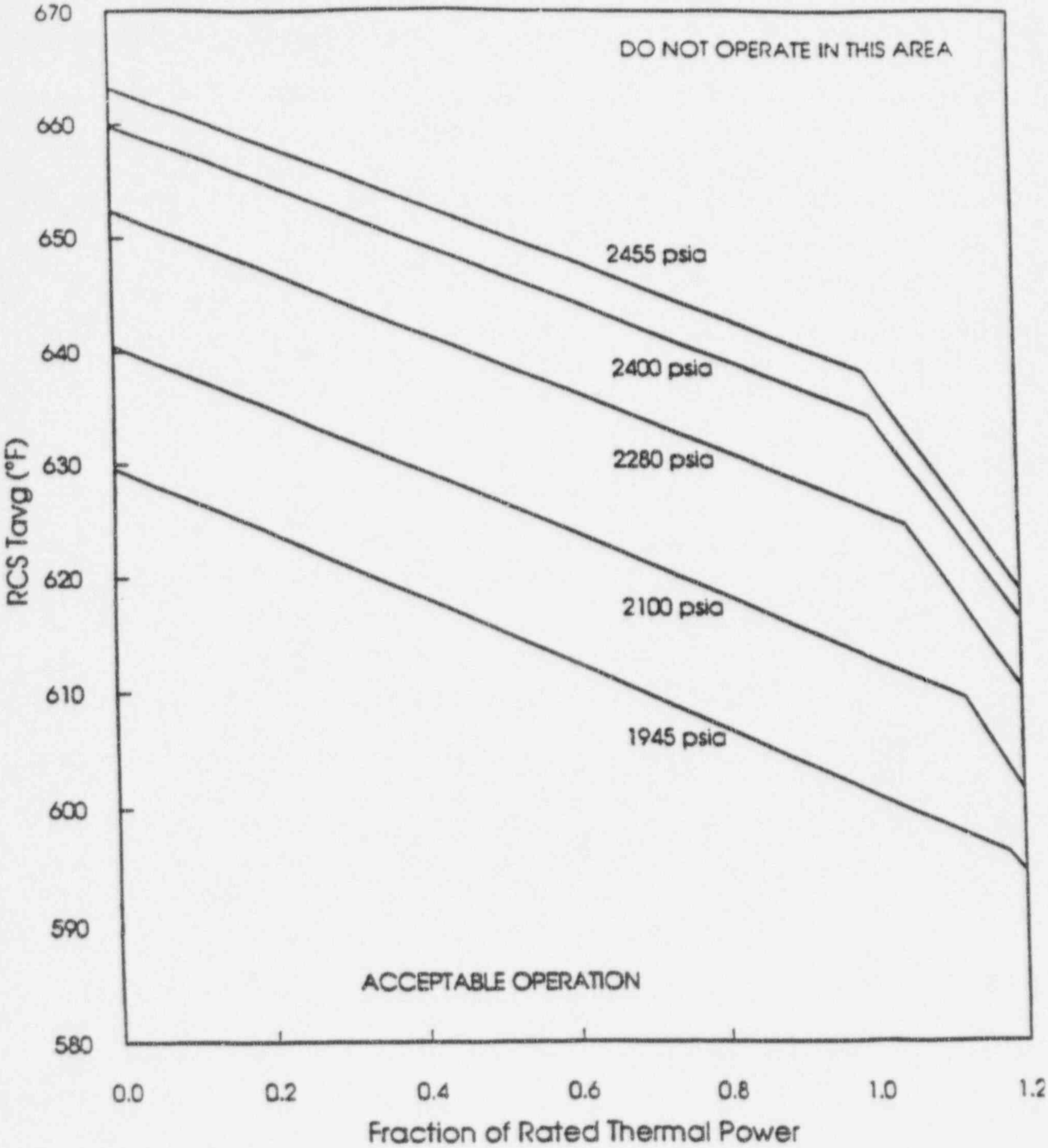

FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - FOUR LOOPS IN OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
5. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
6. Overtemperature ΔT	See Note 1	See Note 3
7. Overpower ΔT	See Note 2	See Note 4
8. Pressurizer Pressure--Low	$\geq$ 1945 psig	$\geq$ 1935 psig
9. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
10. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
11. Low Reactor Coolant Flow	$\geq$ 91% of minimum measured flow per loop*	$\geq$ 90% of minimum measured flow per loop*

*Minimum measured flow is 95,500 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
12. Steam Generator Water Level--Low-Low	≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40% of span at 100% of RATED THERMAL POWER	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing to 39.0% of span at 100% of RATE THERMAL POWER.
13. Undervoltage-Reactor Coolant Pumps	≥ 5082 volts-each bus	≥ 5016 volts-each bus
14. Underfrequency-Reactor Coolant Pumps	≥ 56.4 Hz - each bus	≥ 55.9 Hz - each bus
15. Turbine Trip		
a. Low Trip System Pressure	≥ 45 psig	≥ 42 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
2) P-13 Input	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8, Low Reactor Coolant Loop Flow, and Reactor Coolant Pump Breaker Position	≤ 48% of RATED THERMAL POWER	≤ 49% of RATED THERMAL POWER
d. Low Setpoint Power Range Neutron Flux, P-10, Enable Block of Source Intermediate and Power Range Reactor Trips	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
e. Turbine Impulse Chamber Pressure, P-13, Input to Low Power Reactor Trips Block P-7	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent
18. Reactor Trip Breakers	N.A.	N.A.
19. Automatic Trip and Interlock Logic	N.A.	N.A.

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

NOTE 1 OVERTEMPERATURE ΔT

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

Where:

ΔT = Measured ΔT by Loop Narrow Range RTD,

ΔT_0 = Indicated ΔT at RATED THERMAL POWER,

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

τ_1, τ_2 = Time constants utilized in the lead-lag controller for ΔT, as presented in the Core Operating Limits Report,

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

τ_3 = Time constant utilized in the lag compensator for ΔT, as presented in the Core Operating Limits Report,

K_1 = Overtemperature ΔT reactor trip setpoint as presented in the Core Operating Limits Report,

K_2 = Overtemperature ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report,

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

τ_4, τ_5 = Time constants utilized in the lead-lag controller for T_{avg}, as presented in the Core Operating Limits Report,

T = Average temperature, °F,

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

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

NOTE 1: (Continued)

τ_8 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report,

T' = $\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report,

P = Pressurizer pressure, psig,

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

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

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

(i) for q_t - q_b between the "positive" and "negative" f_1 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of q_t - q_b is more negative than the f_1 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of q_t - q_b is more positive than the f_1 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_1 (ΔI) "positive" slope presented in the Core Operating Limits Report.

NOTE 2: OVERPOWER ΔT

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

Where:

ΔT	=	As defined in Note 1,
ΔT_0	=	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,
K_4	=	Overpower ΔT reactor trip setpoint as presented in the Core Operating Limits Report,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag controller f.. T_{avg} dynamic compensation,
τ_7	=	Time constant utilized in the rate-lag controller for T_{avg}, as presented in the Core Operating Limits Report,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,
K_6	=	Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

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

T	=	As defined in Note 1,
T''	=	$\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,
S	=	As defined in Note 1, and

$f_2(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" $f_2(\Delta I)$ breakpoints as presented in the Core Operating Limits Report; $f_2(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of $q_t - q_b$ is more negative than the $f_2(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than the $f_2(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

NOTE 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.4% of Rated Thermal Power.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% of Rated Thermal Power.

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. This relation 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), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

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

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

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

The curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.50 and a reference cosine axial power shape with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.50\ [1 + (1/RRH)\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER, and RRH is given in the COLR.

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The 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 settings is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the accident analysis to enhance the overall reliability of the Reactor Trip System.

The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance for all trips including those trips assumed in the safety analyses.

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.

Power Range, Neutron Flux (Continued)

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 Positive Rate

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

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.

Overtemperature ΔT

The Overtemperature Delta T 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 response time delays associated with the RTDs mounted in thermowells, 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.

Overpower ΔT

The Overpower Delta T trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under overpower conditions, limits the required range for overtemperature delta T protection, 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, (2) rate of change of temperature for dynamic compensation for instrumentation delays associated with the loop temperature detectors, and (3) axial power distribution, 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 Break."

Pressurizer Pressure

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

Low Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide 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 91% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 91% of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on loss of flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

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

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.5 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.6 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Turbine trip is automatically blocked by P-8 (a power level of approximately 48% of RATED THERMAL POWER with a turbine impulse chamber at approximately 48% of full power equivalent); and on increasing power, reinstated automatically by P-8.

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 Reactor trip and de-energizing of the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

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

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

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

P-13 Provides input to P-7.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 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 when the conditions for the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. 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 maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

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

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

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

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

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% 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 1.0 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

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

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

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*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

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

1) Reactor coolant system boron concentration,

2) Control rod position,

3) Reactor coolant system average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% 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 within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0.

APPLICABILITY: Figure 3.1-0 and COLR Figure 1 Limits - MODES 1 and 2* only.#
End of Cycle Life (EOL) Limit - MODES 1, 2, and 3 only.#

ACTION:

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

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

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

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

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

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*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

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

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

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

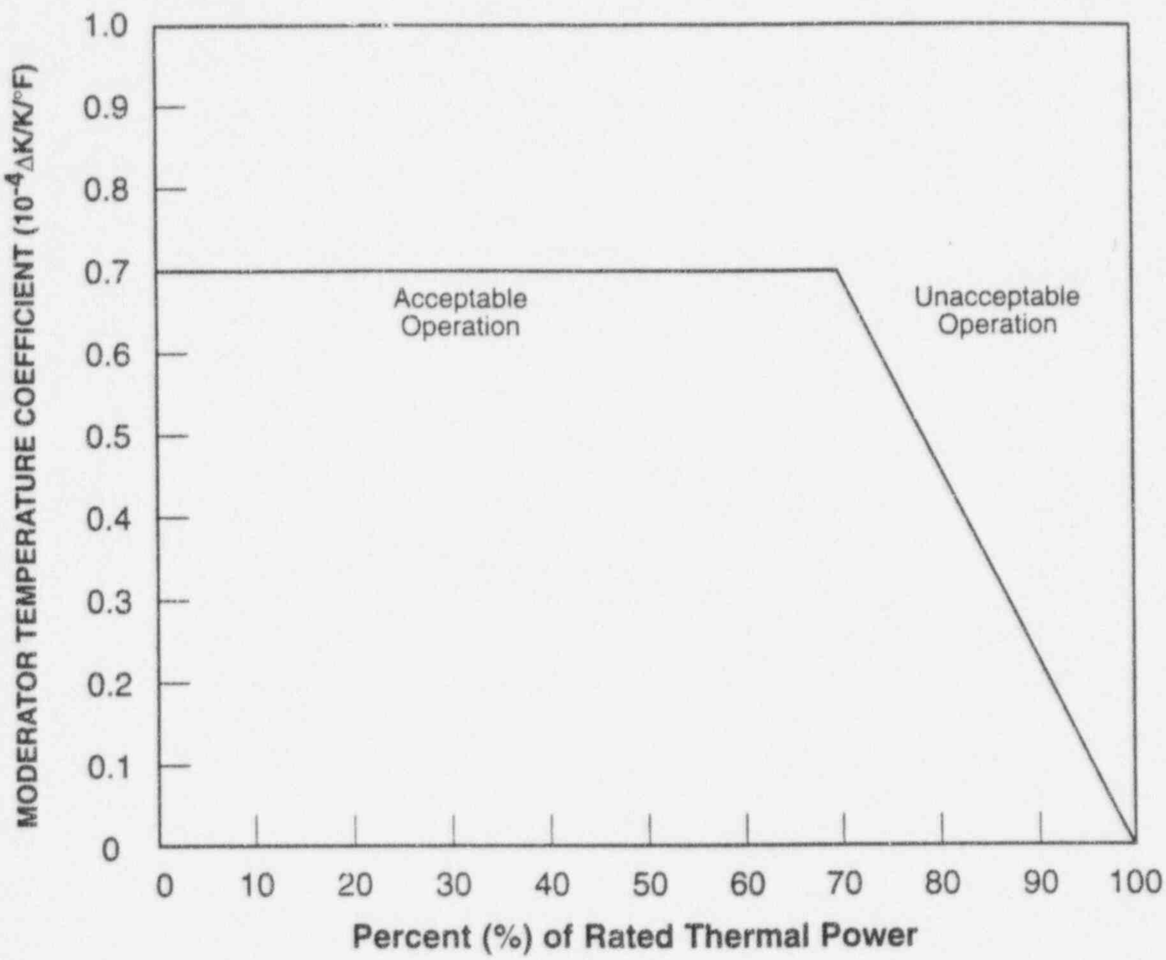

FIGURE 3.1-0
MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2#*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

#With K_{eff} greater than or equal to 1.0.

*See Special Test Exception 3.10.3.

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

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°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 a differential pressure across the pump of greater than or equal to 2380 psid is developed when tested pursuant to Specification 4.0.5.

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

#Two charging pumps may be operable and operating for ≤15 minutes to allow swapping charging pumps.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two# charging pumps shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.1.2.4.2 All centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each charging pump has been isolated from the Reactor Coolant System by at least two isolation valves with power removed from the valve operators.

#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. Two charging pumps may be operable and operating for ≤15 minutes to allow swapping charging pumps.

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 and at least one associated Heat Tracing System with:

   1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

   2) A minimum boron concentration as presented in the Core Operating Limits Report, and

   3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

   1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

   2) A minimum boron concentration as presented in the Core Operating Limits Report, and

   3) A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

   1) Verifying the boron concentration of the water,

   2) Verifying the contained borated water volume, and

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

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System and at least one associated Heat Tracing System with:

   1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

   2) A minimum boron concentration as presented in the Core Operating Limits Report, and

   3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

   1) A minimum contained borated water volume as presented in the Core Operating Limits Report or Specification 3.5.5a, whichever is larger,

   2) A minimum boron concentration as presented in the Core Operating Limits Report,

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

   4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

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

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

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

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

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

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

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. 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;

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*See Special Test Exceptions 3.10.2 and 3.10.3.

ACTION (Continued)

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

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

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

d. With more than one full-length 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 Specification 3.1.3.6. 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.

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 Misoperation

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

Major Secondary System Pipe Rupture

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

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 18 months. The Reactor Trip System Breakers can be closed in order to perform this surveillance.

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

#See Special Test Exception 3.10.5.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.2 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 (*) of RATED THERMAL POWER.

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.

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

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 12 hours thereafter.

---

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.0.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits specified in the COLR, 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 insertion limits specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

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

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*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.0.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the acceptable limits as specified in the Core Operating Limits Report (COLR).

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

ACTION:

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

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

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

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   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 Monitoring Alarm to OPERABLE status.

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

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

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*See Special Test Exception 3.10.2.

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(X,Y,Z)$ shall be limited by imposing the following relationship:

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

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

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

$K(Z)$ = the normalized $F_Q(X,Y,Z)$ limit specified in the COLR for the appropriate fuel type, and

$F_Q^{MA}(X,Y,Z)$ = the measured heat flux hot channel factor $F_Q^M(X,Y,Z)$ with the adjustments specified in 4.2.2.3

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(X,Y,Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours, and

b. Control the AFD to within new AFD limits which are determined by reducing the allowable power at each point along the AFD limit lines of Specification 3.2.1 at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and reset the AFD alarm setpoints to the modified limits within 8 hours, and

c. POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit, and

d. 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 $F_Q(X,Y,Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q^M(X,Y,Z)$[1] shall be evaluated to determine whether $F_Q(X,Y,Z)$ 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. Measuring $F_Q^M(X,Y,Z)$ at the earliest of:

1. At least once per 31 Effective Full Power Days, or

2. Upon reaching equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last determined[2], or

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

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(1) No additional uncertainties are required in the following equations for $F_Q^M(X,Y,Z)$ because the limits include uncertainties.

(2) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

c. Performing the following calculations:

1. For each core location, calculate the % margin to the maximum allowable design as follows:

$$\% \text{ Operational Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{OP}}\right) \times 100\%$$

$$\% \text{ RPS Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{RPS}}\right) \times 100\ \%$$

where $[F_Q^L(X,Y,Z)]^{OP}$ and $[F_Q^L(X,Y,Z)]^{RPS}$ are the Operational and RPS design peaking limits defined in the COLR.

2. Find the minimum Operational Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then either of the following actions shall be taken:

(a) Within 15 minutes:

(1) Control the AFD to within new AFD limits that are determined by:

$$(\text{AFD Limit})_{\text{negative}}^{\text{reduced}} = (\text{AFD Limit})_{\text{negative}}^{\text{COLR}^{(3)}} - \text{MARGIN}_{\text{OP}}^{\text{MIN}}$$

$$(\text{AFD Limit})_{\text{positive}}^{\text{reduced}} = (\text{AFD Limit})_{\text{positive}}^{\text{COLR}^{(3)}} - \text{MARGIN}_{\text{OP}}^{\text{MIN}}$$

where $\text{MARGIN}_{\text{OP}}^{\text{MIN}}$ is the minimum margin from 4.2.2.2.c.1, and

(2) Within 8 hours, reset the AFD alarm setpoints to the modified limits of 4.2.2.2.c.2.a, or

(b) Comply with the ACTION requirements of Specification 3.2.2, treating the margin violation in 4.2.2.2.c.1 above as the amount by which F_Q^{MA} is exceeding its limit.

---

(3) Defined and specified in the COLR per Specification 6.9.1.9.

SURVEILLANCE REQUIREMENTS (Continued)

3. Find the minimum RPS Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then the following action shall be taken:

Within 72 hours, reduce the K_1 value for OTΔT by:

$$K_1 \text{ adjusted} = K_1^{(4)} - [\text{KSLOPE}^{(3)} \times \text{Margin}\ {}^{\text{min}}_{\text{RPS}}] \text{ absolute value}$$

where MARGIN $^{\text{min}}_{\text{RPS}}$ is the minimum margin from 4.2.2.2.c.1.

d. Extrapolating[5] at least two measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$$[F_Q^M(X,Y,Z)] \text{ (extrapolated)} \geq [F_Q^L(X,Y,Z)]^{OP} \text{ (extrapolated), and}$$

$$\frac{[F_Q^M(X,Y,Z)] \text{ (extrapolated)}}{[F_Q^L(X,Y,Z)]^{OP} \text{ (extrapolated)}} > \frac{[F_Q^M(X,Y,Z)]}{[F_Q^L(X,Y,Z)]^{OP}}$$

or

$$[F_Q^M(X,Y,Z)] \text{ (extrapolated)} \geq [F_Q^L(X,Y,Z)]^{RPS} \text{ (extrapolated), and}$$

$$\frac{[F_Q^M(X,Y,Z)] \text{ (extrapolated)}}{[F_Q^L(X,Y,Z)]^{RPS} \text{ (extrapolated)}} > \frac{[F_Q^M(X,Y,Z)]}{[F_Q^L(X,Y,Z)]^{RPS}}$$

either of the following actions shall be taken:

1. $F_Q^M(X,Y,Z)$ shall be increased by 2 percent over that specified in 4.2.2.2.a, and the calculations of 4.2.2.2.c repeated, or

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(3) Defined and specified in the COLR per Specification 6.9.1.9.

(4) K_1 value from Table 2.2-1.

(5) Extrapolation of F_Q^M for the initial flux map taken after reaching equilibrium conditions is not required since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of F_Q^M limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

SURVEILLANCE REQUIREMENTS (Continued)

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.2.2.c.1 shall be performed no later than the time at which the margin in 4.2.2.2.c.1 is extrapolated to be equal to zero.

e. The limits in Specifications 4.2.2.2.c and 4.2.2.2.d are not applicable in the following core plane 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.

4.2.2.3 When a full core power distribution map is taken for reasons other than meeting the requirements of Specification 4.2.2.2, an overall $F_Q^M(X,Y,Z)$ shall be determined, then increased by 3% to account for manufacturing tolerances, further increased by 5% to account for measurement uncertainty, and further increased by the radial-local peaking factor to obtain a maximum local peak. This value shall be compared to the limit in Specification 3.2.2.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}(X,Y)$ shall be limited by imposing the following relationship:

$$F_{\Delta H}{}^{M}(X,Y) \leq [F_{\Delta H}{}^{L}(X,Y)]^{LCO}$$

where: $F_{\Delta H}{}^{M}(X,Y)$ - the measured radial peak.

$[F_{\Delta H}{}^{L}(X,Y)]^{LCO}$ - the maximum allowable radial peak as defined in Core Operating Limits Report (COLR).

APPLICABILITY: MODE 1.

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding its limit:

a. Within 2 hours, reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH%[1] for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds the limit, and

b. Within 6 hours either:

   1. Restore $F_{\Delta H}{}^{M}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH% for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds that limit, and

c. Within 72 hours of initially being outside the limit of Specification 3.2.3, either:

   1. Restore $F_{\Delta H}{}^{M}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Perform the following actions:

---

(1) RRH is the amount of THERMAL POWER reduction required to compensate for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

LIMITING CONDITION FOR OPERATION

ACTION:

(a) Reduce the OTΔT K_1 term in Table 2.2-1 by at least TRH[2] for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds the limit, and

(b) Verify through incore mapping that $F_{\Delta H}{}^{M}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a., or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

---

[2] TRH is the amount of OTΔT K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

d. 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. and/or c.2 above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}{}^{M}(X,Y)$ is demonstrated, through incore flux mapping, to be within the Limit specified in the COLR prior to exceeding the following THERMAL POWER levels:

   1. 50% of RATED THERMAL POWER,

   2. 75% of RATED THERMAL POWER, and

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F_{\Delta H}{}^{M}(X,Y)$ shall be evaluated to determine whether $F_{\Delta H}(X,Y)$ 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. Measuring $F_{\Delta H}{}^{M}(X,Y)$ according to the following schedule:

   1. Upon reaching equilibrium conditions after exceeding 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_{\Delta H}{}^{M}(X,Y)$ was last determined[3], or

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

   3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

c. Performing the following calculations:

   1. For each location, calculate the % margin to the maximum allowable design as follows:

---

(3) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

$$\% \; F_{\Delta H} \text{ Margin} = \left(1 - \frac{F_{\Delta H}{}^{M}(X,Y)}{[F_{\Delta H}{}^{L}(X,Y)]^{surv}}\right) \times 100\%$$

No additional uncertainties are required for $F_{\Delta H}{}^{M}(X,Y)$, because $[F_{\Delta H}{}^{L}(X,Y)]^{surv}$ includes uncertainties.

2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above. If any margin is less than zero, comply with the ACTION requirements of Specification 3.2.3 as if $[F_{\Delta H}{}^{L}(X,Y)]^{surv}$ is the same as $F_{\Delta H}{}^{L}(X,Y)]^{LCO}$.

d. Extrapolating[4] at least two measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$F_{\Delta H}{}^{M}(X,Y)$ (extrapolated) $\geq [F_{\Delta H}{}^{L}(X,Y)^{surv}$ (extrapolated) and

$$\frac{F_{\Delta H}{}^{M}(X,Y) \quad \text{(extrapolated)}}{[F_{\Delta H}{}^{L}(X,Y)]^{surv} \quad \text{(extrapolated)}} > \frac{F_{\Delta H}{}^{M}(X,Y)}{[F_{\Delta H}{}^{L}(X,Y)]^{surv}}$$

either of the following actions shall be taken:

1. $F_{\Delta H}{}^{M}(X,Y)$ shall be increased by 2 percent over that specified in 4.2.3.2.a, and the calculations of 4.2.3.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.3.2.c shall be performed no later than the time at which the margin in 4.2.3.2.c is extrapolated to be equal to zero.

---

(4) Extrapolation of $F_{\Delta H}{}^{M}$ for the initial flux map taken after reaching equilibrium conditions is not required since the initial flux map establishes the baseline measurement for future trending.

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 1.02 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 Exception 3.10.2.

**Not applicable until calibration of the excore detectors is completed subsequent to refueling.

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 1.02, 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 a 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},

b. Pressurizer Pressure, and

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1.

ACTION:

a. With either of the parameters identified in 3.2.5a. and b. above 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.

b. With the combination of Reactor Coolant System total flow rate and THERMAL POWER within the region of restricted operation specified on Figure 3.2.1, within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by Figure 3.2-1.

c. With the combination of RCS total flow rate and THERMAL POWER within the region of prohibited operation specified on Figure 3.2-1:

1. Within 2 hours either:

a) Restore the combination of RCS total flow rate and THERMAL POWER to within the region of permissible operation,

b) Restore the combination of Reactor Coolant System total flow rate and THERMAL POWER to within the region of restricted operation and comply with action b. above, or

c) 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.

2. Within 24 hours of initially being within the region of prohibited operation specified in Figure 3.2-1, verify that the combination of THERMAL POWER and RCS total flow rate are restored to within the regions of permissible or restricted operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

3/4.2.5 DNB PARAMETERS

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be measured by averaging the indications (meter or computer) of the operable channels and verified to be within their limits at least once per 12 hours.

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

4.2.5.3 The RCS total flow rate shall be determined by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	INDICATION	# OPERABLE CHANNELS	LIMITS*
Indicated Reactor Coolant System T_{avg}	meter	4	≤ 590.5°F
	meter	3	≤ 590.2°F
	computer	4	≤ 591.0°F
	computer	3	≤ 590.8°F
Indicated Pressurizer Pressure**	meter	4	≥ 2226.5 psig
	meter	3	≥ 2229.8 psig
	computer	4	≥ 2221.7 psig
	computer	3	≥ 2224.2 psig
Reactor Coolant System Total Flow Rate			Figure 3.2-1

*Limits applicable during four-loop operation.

**Limits 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% RATED THERMAL POWER.

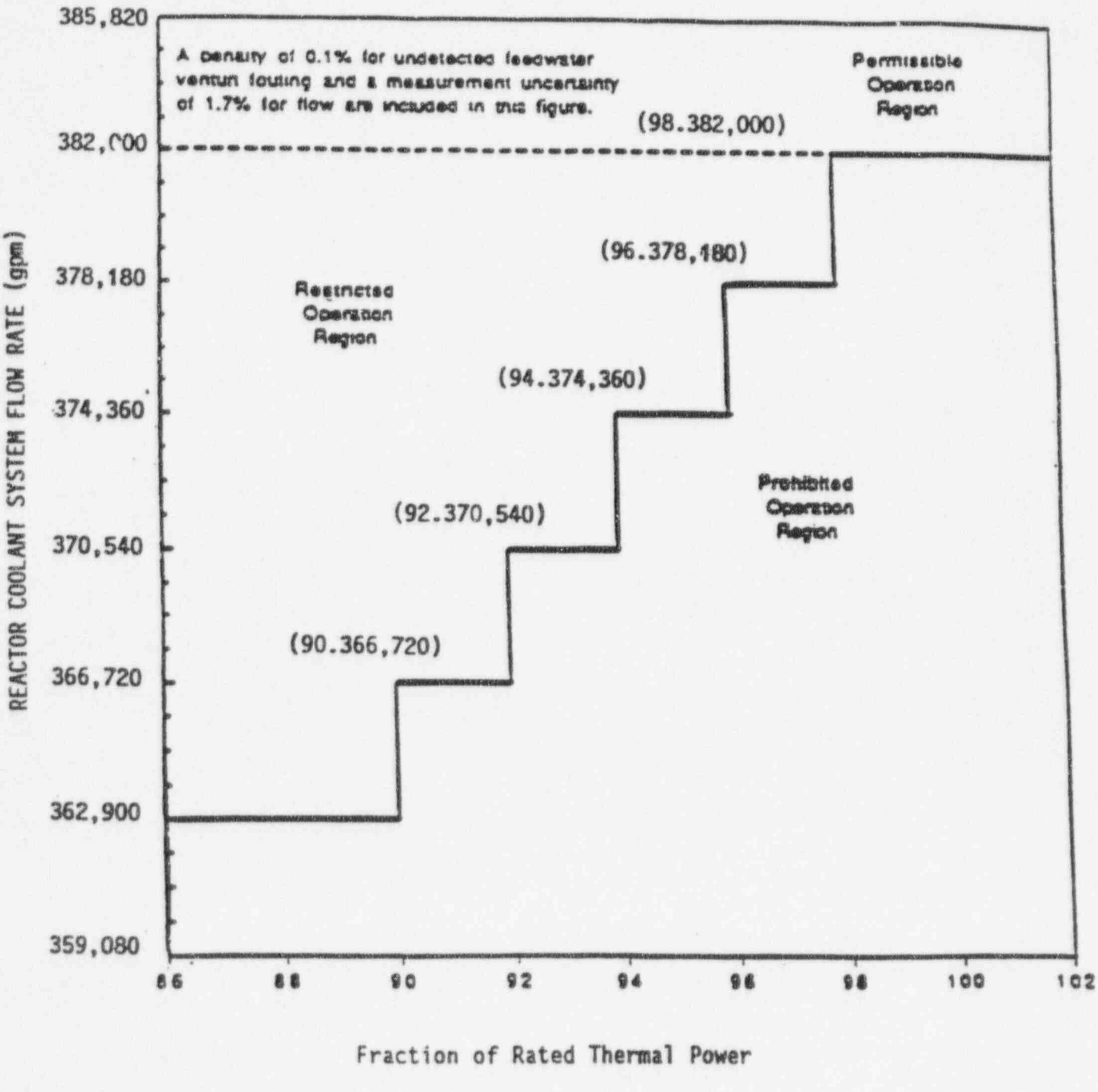

FIGURE 3.2-1. REACTOR COOLANT SYSTEM TOTAL FLOW RATE VERSUS RATED THERMAL POWER - FOUR LOOPS IN OPERATION

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System Instrumentation channel and interlock 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.

4.3.1.3 The response time of RTDs associated with the Reactor Trip System shall be demonstrated to be within their limits (see note 2 to Table 3.3-2) at least once per 18 months.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux - High Setpoint	4	2	3	1, 2	2
Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
5. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
c. Shutdown	2	0	1	3, 4, and 5	5
6. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	6
Three Loop Operation	(**)	(**)	(**)	(**)	(**)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	6
Three Loop Operation	(**)	(**)	(**)	(**)	(**)
8. Pressurizer Pressure-Low	4	2	3	1	6
9. Pressurizer Pressure--High	4	2	3	1, 2	6
10. Pressurizer Water Level--High	3	2	2	1	6
11. Low Reactor Coolant Flow					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6
12. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Undervoltage-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6
14. Underfrequency-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6
15. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1	6
b. Turbine Stop Valve Closure	4	4	1	1	11
16. Safety Injection Input from ESF	2	1	2	1, 2	7
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
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Low Setpoint Power Range Neutron Flux, P-10	4	2	3	1, 2	8
e. Turbine Impulse Chamber Pressure, P-13	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
18. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10
19. Automatic Trip and Interlock Logic	2	1	2	1, 2	7
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the Reactor Trip System breakers in the closed position, the Control Rod Drive System capable of rod withdrawal.

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

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

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

ACTION STATEMENTS

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, 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, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

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

a. The inoperable channel is placed in the tripped condition within 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 and Specification 4.3.2.1.

ACTION 7 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

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 the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

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

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 12 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 9. 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	≤ 0.5 second (1)
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Intermediate Range, Neutron Flux	N.A.
5. Source Range, Neutron Flux	N.A.
6. Overtemperature ΔT	≤ 10.0 seconds (1)(2)
7. Overpower ΔT	≤ 10.0 seconds (1)(2)
8. Pressurizer Pressure--Low	≤ 2.0 seconds
9. Pressurizer Pressure--High	≤ 2.0 seconds
10. Pressurizer Water Level--High	N.A.

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

(2) The ≤ 10.0 second response time includes a 6.5 second delay for the RTDs mounted in thermowells.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	≤ 1.0 second
b. Two Loops (Above P-7 and below P-8)	≤ 1.0 second
12. Steam Generator Water Level--Low-Low	≤ 3.5 seconds
13. Undervoltage-Reactor Coolant Pumps	< 1.5 seconds
14. Underfrequency-Reactor Coolant Pumps	< 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.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R (11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1###, 2
5. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),Q(9)	N.A.	N.A.	2##, 3, 4, 5
6. Overtemperature ΔT	S	R(15)	Q	N.A.	N.A.	1, 2
7. Overpower ΔT	S	R(15)	Q	N.A.	N.A.	1, 2
8. Pressurizer Pressure--Low	S	R	Q	N.A.	N.A.	1
9. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2
10. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1
11. Low Reactor Coolant Flow	S	R	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level--Low-Low	S	R	Q	N.A.	N.A.	1, 2
13. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
14. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q	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)	N.A	N.A.	N.A.	2##
b. Power Range Neutron Flux, P-8	N.A.	R(4)	N.A	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
c. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	N.A	N.A.	N.A.	1, 2
d. Turbine Impulse Chamber Pressure, P-13	N.A.	R	N.A.	N.A.	N.A.	1
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 12)	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. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	M(13),R(14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With the Reactor Trip System breakers closed and the Control Rod Drive System capable of rod withdrawal.

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

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

(1) - If not performed in previous 31 days.

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

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

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

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

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

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

(8) - Deleted.

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

(10) - Setpoint verification is not required.

TABLE 4.3-1 (Continued)

TABLE NOTATION

(11) - The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function.

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

(13) - Prior to placing breaker in service, a local manual shunt trip shall be performed.

(14) - The automatic undervoltage trip capability shall be verified operable.

(15) - Overtemperature setpoint, overpower setpoint, and T_{avg} channels require an 18 month channel calibration. Calibration of the ΔT channels is required at the beginning of each cycle upon completion of the precision heat balance. RCS loop ΔT values shall be determined by precision heat balance measurements at the beginning of each cycle.

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

b. With an ESFAS Instrumentation channel 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 the 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, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3	15
d. Pressurizer Pressure - Low-Low	4	2	3	1, 2, 3#	19
e. Steam Line Pressure-Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (continued)					
b. Phase "B" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	4	2	3	1, 2, 3	16
c. Purge and Exhaust Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) System	2	1	2	1, 2, 3	22
2) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
d. Negative Steam Line Pressure Rate - High					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	3##	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)
e. Steam Line Pressure - Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

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 & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relay	2	1	2	1, 2	21
b. Steam Generator Water Level--High-High	3/stm. gen.	2/stm. gen. in any operating stm gen.	2/stm. gen. in each operating stm. gen.	1, 2	15
c. Doghouse Water Level (Feedwater Isolation Only)	3/train/ Doghouse	2/train/ Doghouse	2/train/ Doghouse	1, 2	25
6. Containment Pressure Control System	8(4/train)	4/train	8	1, 2, 3, 4	26

<u>TABLE 3.3-3 (Continued)</u>
<u>ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION</u>

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater					
a. Manual Initiation	2	1	2	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Stm. Gen. Water Level-Low-Low					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen in each operating stm. gen	1, 2, 3	19
d. Auxiliary Feedwater Suction Pressure - Low	2/motor driven pump	2/pump	2 of the same train/pump	1,2,3	24
(Suction Supply Automatic Realignment)	4/turbine driven pump	2/pump	2 of the same train/pump	1,2,3	24
e. Safety Injection Start Motor-Driven Pumps					

See Item 1. above for all Safety Injection initiating functions and requirements

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater (continued)					
f. Station Blackout (Note 1) Start Motor-Driven Pumps and Turbine-Driven Pump					
1) 4 kV Loss of Voltage	3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19
2) 4 kV Degraded Voltage	3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19
g. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	2-1/MFWP	2-1/MFWP	2-1/MFWP	1, 2#	27
8. Automatic Switchover to Recirculation					
RWST Level	3	2	2	1, 2, 3	15b
9. Loss of Power					
a. 4 kV Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15a
b. 4 kV Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15a
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22
d. Steam Generator Level, P-14	3/stm gen.	2/stm gen. in any operating stm gen.	2/stm gen. in each operating stm gen.	1, 2, 3	20

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

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

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

Note 1: Turbine driven auxiliary feedwater pump will not start on a blackout signal coincident with a safety injection signal.

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 12 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 15a With the number of OPERABLE channels less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours. With more than one channel inoperable, enter Specification 3.8.1.1.

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

ACTION 16 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

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

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

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

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

a. The inoperable channel is placed in the tripped condition within 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 and Specification 4.3.2.1.

ACTION 20 - 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 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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

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

ACTION 24 - With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.7.1.2. With the channels associated with more than one auxiliary feedwater pump inoperable, immediately declare the associated auxiliary feedwater pumps inoperable and take the action required by Specification 3.7.1.2.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 25 - With one of the two trains of doghouse water level instrumentation inoperable (less than the minimum required number of channels operable), restore the inoperable train to operable status in 72 hours. After 72 hours with one train inoperable, or within one hour with 2 trains inoperable, monitor doghouse water level in the affected doghouse continuously until both trains are restored to operable status.

ACTION 26 - With any of the eight channels inoperable, place the inoperable channel(s) in the start permissive mode within one hour and apply the applicable action statement (Containment Spray - T.S. 3.6.2, Containment Air Return/Hydrogen Skimmer - T.S. 3.6.5.6).

ACTION 27 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water.		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High	≤ 1.1 psig	≤ 1.2 psig
d. Pressurizer Pressure--Low-Low	≥ 1845 psig	≥ 1835 psig
e. Steam Line Pressure - Low	≥ 775 psig	≥ 755 psig
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	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.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
c. Purge and Exhaust Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig
d. Negative Steam Line Pressure Rate - High	≤ 100 psi with a rate/lag function time constant ≥ 50 seconds	≤ 120 psi with a rate/lag function time constant ≥ 50 seconds
e. Steam Line Pressure - Low	≥ 775 psig	≥ 755 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	≤ 82% of narrow range instrument span each steam generator	≤ 83% of narrow range instrument span each steam generator
c. Doghouse Water Level-High (Feedwater Isolation Only)	12"	13"
6. Containment Pressure Control System		
Start Permissive/Termination (SP/T)	0.3 ≤ SP/T ≤ 0.4 PSIG	0.25 ≤ SP/T ≤ 0.45 PSIG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40.0% of span at 100% of RATED THERMAL POWER.	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 39.0% of span at 100% of RATED THERMAL POWER.
2) Start Turbine-Driven Pumps	12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40.0% of span at 100% of RATED THERMAL POWER.	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 39.0% of span at 100% of RATED THERMAL POWER.
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	≥ 2 psig	≥ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TABLE 3.3-4 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater (continued)		
f. Station Blackout - Start Motor-Driven Pumps and Turbine-Driven Pump (Note 1)		
1) 4 kV Loss of Voltage	3157 ± 45 volts with a 8.5 ± 0.5 second time delay	≥ 3108 volts
2) 4 kV Degraded Voltage	≥ 3703 volts with ≤ 11 second with SI and ≤ 600 second without SI time delays	≥ 3685.5 volts
g. Trip of Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.
8. Automatic Switchover to Recirculation RWST Level	≥ 90 inches	≥ 80 inches
9. Loss of Power		
a. 4 kV Loss of Voltage	3157 ± 45 volts with a 8.5 ± 0.5 second time delay	≥ 3108 volts
b. 4 kV Degraded Voltage	≥ 3703 volts with ≤ 11 second with SI and ≤ 600 second without SI time delays	≥ 3685.5 volts
10. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	≤ 1955 psig	≤ 1965 psig
b. T_{avg}, P-12	≥ 553°F	≥ 551°F
c. Reactor Trip, P-4	N.A.	N.A.
d. Steam Generator Level, P-14	See Item 5b. above for all Trip Setpoints and Allowable Values.	

Note 1: The turbine driven pump will not start on a blackout signal coincident with a safety injection signal.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Containment Isolation	
Phase "A" Isolation	N.A.
Phase "B" Isolation	N.A.
Purge and Exhaust Isolation	N.A.
d. Steam Line Isolation	N.A.
e. Feedwater Isolation	N.A.
f. Auxiliary Feedwater	N.A.
g. Nuclear Service Water	N.A.
h. Component Cooling Water	N.A.
i. Reactor Trip (from SI)	N.A.
j. Start Diesel Generators	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27^{(1)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
i. Start Diesel Generators	≤ 11

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low-Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water System	$\leq 76^{(1)}/65^{(3)}$
h. Component Cooling Water	$\leq 76^{(1)}/65^{(3)}$
i. Start Diesel Generators	≤ 11
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12^{(3)}/22^{(4)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Steam Line Isolation	≤ 10
i. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
j. Start Diesel Generators	≤ 11

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
5. Containment Pressure-High-High	
a. Containment Spray	≤ 45
b. Containment Isolation-Phase "B"	N.A.
c. Steam Line Isolation	≤ 10
6. Steam Generator Water Level-High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ 12
7. Steam Generator Water Level LowLow	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-driven Auxiliary Feedwater Pumps	≤ 60
8. Negative Steam Line Pressure Rate - High	
Steam Line Isolation	≤ 10
9. Start Permissive	
Containment Pressure Control System	N.A.
10. Termination	
Containment Pressure Control System	N.A.
11. Auxiliary Feedwater Suction Pressure - Low	
Auxiliary Feedwater Pumps (Suction Supply Automatic Realignment)	≤ 13
12. RWST Level	
Automatic Switchover to Recirculation	≤ 60

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
13. Station Blackout	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine-Driven Auxiliary Feedwater Pump [(6)]	≤ 60
14. Trip of Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
15. Loss of Power	
a. 4 kV Loss of Voltage	≤ 11
b. 4 kV Degraded Voltage	≤ 11 with SI, and ≤ 600 without SI

TABLE 3.3-5 (Continued)

TABLE NOTATION

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

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

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

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

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

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

(6) The turbine driven pump does not start on a blackout signal coincident with a safety injection signal.

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, Component Cooling Water, Start Diesel Generators, and Nuclear 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	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
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-High	S	R	Q	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 and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
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.						

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								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Negative Steam Line Pressure Rate-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	3
e. Steam Line Pressure--Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	Q	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Doghouse Water Level-High (Feedwater Isolation Only)	S	N.A	N.A	R	N.A.	N.A.	N.A.	1, 2
6. Containment Pressure Control System Start Permissive/Termination	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	Q	N.A.	N.A.	N.A	N.A.	1, 2, 3
d. Auxiliary Feedwater Suction Pressure-Low	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
f. Station Blackout	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
8. Automatic Switchover to Recirculation								
RWST Level	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
9. Loss of Power								
a. 4 kV Loss of Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3, 4
b. 4 kV Degraded Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low, Low T_{avg}, P-12	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Level, P-14	See Item 5b for all surveillance requirements.							

TABLE NOTATION

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

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 exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the action shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

MONITOR	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment Atmosphere Gaseous Radioactivity-High (Low Range-EMF-39)	1	1	1, 2, 3, 4	***	26
2. Spent Fuel Pool Radioactivity-High (EMF-42)	1	1	**	$\leq 1.7 \times 10^{-4}$ μCi/ml	30
3. Criticality-Radiation Level (2EMF-4)	1	1	*	$\leq$ 15 mR/hr	28
4. Gaseous Radioactivity-RCS Leakage Detection (Low Range - EMF-39)	N.A.	1	1, 2, 3, 4	N.A.	29
5. Particulate Radioactivity-RCS Leakage Detection (Low Range - EMF-38)	N.A.	1	1, 2, 3, 4	N.A.	29

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

MONITOR	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
6. Control Room Air Intake Radioactivity-High (EMF-43a and 43b)	1 per station	2 per station	All	$\leq 3.4 \times 10^{-4}$ μCi/ml	27

TABLE NOTATION

* - With fuel in the fuel storage areas or fuel building.

** - With irradiated fuel in the fuel storage areas or fuel building.

*** - Must satisfy the requirements of McGuire Selected Licensee Commitment 16.11-6.

ACTION STATEMENTS

ACTION 26 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge 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 Ventilation System outside air intake which contains the inoperable instrumentation.

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 pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel building.

ACTION 29 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 30 - With less than the minimum channels OPERABLE requirement, operation may continue provided the Fuel Handling Ventilation Exhaust System requirements of Specification 3/4.9.11 are met.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

MONITOR	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	ANALOG MODES REQUIRING SURVEILLANCE
1. Containment Atmosphere Gaseous Radioactivity-High (Low Range-EMF-39)	S	R	Q	1, 2, 3, 4
2. Spent Fuel Pool Ventilation Radioactivity-High (EMF-42)	S	R	Q	**
3. Criticality-High Radiation Level (2EMF-4)	S	R	Q	*
4. Gaseous Radioactivity-RCS Leakage Detection (Low Range-EMF-39)	S	R	Q	1, 2, 3, 4
5. Particulate Radioactivity RCS Leakage Detection (Low Range-EMF-38)	S	R	Q	1, 2, 3, 4
6. Control Room Air Intake-Radioactivity High (EMF-43a and EMF-43b)	S	R	Q	All

TABLE NOTATION

* - With fuel in the fuel handling area.

** - With irradiated fuel in the fuel handling area.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 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,

b. Monitoring the QUADRANT POWER TILT RATIO, or

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

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

b. Monitoring the QUADRANT POWER TILT RATIO, or

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

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

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

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required 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 SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status according to b.1 or b.2 below:

   b.1 Instruments 1-15: within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

   b.2 Instruments 16 and 17: according to Technical Specification 3.7.4.a.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Temperature - T_{HOT} and T_{COLD} (Wide Range)	$2/T_{HOT}$ $2/T_{COLD}$	$1/T_{HOT}$ $1/T_{COLD}$
3. Reactor Coolant Pressure - Wide Range	2	1
4. Pressurizer Water Level	2	1
5. Steam Line Pressure	2/steam generator	1/steam generator
6. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
7. Refueling Water Storage Tank Water Level	2	1
8. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator
9. Reactor Coolant System Subcooling Margin Monitor	2	1
10. Containment Water Level (Wide Range)	2	1
11. In Core Thermocouples	4/core quadrant	2/core quadrant
12. Containment Atmosphere - High Range Monitor (EMF-51a or 51b)	1	1
13. Reactor Vessel Level Instrumentation		
a. Dynamic Head (D/P) Range	2	1
b. Lower Range	2	1
14. Neutron Flux - Wide Range	2	1
15. Containment Hydrogen Concentration	2	1
16. Diesel Generator Cooling Water Heat Exchanger RN Flow*	1/diesel generator	1/diesel generator
17. Containment Spray Heat Exchanger RN Flow*	1/train	1/train

*Not applicable if the associated outlet valve is set to its flow balance position with power removed or if the associated outlet valve's flow balance position is fully open.

TABLE 4.3-7
ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Temperature - T_{HOT} and T_{COLD} (Wide Range)	M	R
3. Reactor Coolant Pressure - Wide Range	M	R
4. Pressurizer Water Level	M	R
5. Steam Line Pressure	M	R
6. Steam Generator Water Level - Narrow Range	M	R
7. Refueling Water Storage Tank Water Level	M	R
8. Auxiliary Feedwater Flow Rate	M	R
9. Reactor Coolant System Subcooling Margin Monitor	M	R
10. Containment Water Level (Wide Range)	M	R
11. In Core Thermocouples	M	R
12. Containment Atmosphere - High Range Monitor (EMF-51a or 51b)	M	R
13. Reactor Vessel Level Instrumentation		
a. Dynamic Head (D/P) Range	M	R
b. Lower Range	M	R
14. Neutron Flux - Wide Range	M	R
15. Containment Hydrogen Concentration	M	R
16. Diesel Generator Cooling Water Heat Exchanger RN Flow	M	R
17. Containment Spray Heat Exchanger RN Flow	M	R

3/4 3.3.7 DELETED

3/4 3.3.8 DELETED

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 3.3-13.

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-13

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Not Used			
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitor	1 per station	**	1
b. Oxygen Monitors	2 per station	**	2
3. Not Used			
4. Not Used			

**During WASTE GAS HOLDUP SYSTEM operation.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

ACTION 2 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With two channels inoperable, be in at least HOT STANDBY within 6 hours.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

MONITOR	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES REQUIRING SURVEILLANCE
1. Not Used				
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System				
a. Hydrogen Monitor	D	Q(1)	M	**
b. Oxygen Monitor	D	Q(2)	M	**
c. Oxygen Monitor (alternate)	D	Q(2)	M	**
3. Not Used				
4. Not Used				

TABLE NOTATION

**During WASTE GAS HOLDUP SYSTEM operation.

(1) The CHANNEL CALIBRATION shall include the use of standard gas samples corresponding to alarm setpoints in accordance with the manufacturer's recommendations.

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

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

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

SURVEILLANCE REQUIREMENT

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

---

*See Special Test Exception 3.10.4.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

APPLICABILITY: MODE 3

ACTION:

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

b. With less than the above required reactor coolant loops in operation, restore the required loops to operation within 72 hours or open the Reactor Trip System breakers.

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

SURVEILLANCE REQUIREMENT

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

4.4.1.2.2 At least the above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

e. RHR Loop A, and

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

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

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

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*A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless: (1) the pressurizer water volume is less than 92% (1600 cubic feet), or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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

SURVEILLANCE REQUIREMENT

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

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

4.4.1.3.3 At least once 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 12%.

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

ACTION:

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

b. With no RHR 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 REQUIREMENT

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.

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

#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 300°F unless: (1) the pressurizer water level is less than 92% (1600 cubic feet), or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperature.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required 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 REQUIREMENT

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

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig + 3%, - 2%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

*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 level of less than or equal to 92% (1600 cubic feet), and at least two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

c. With two PORVs inoperable due to causes other than excessive leakage, within 1 hour either restore the PORVs to OPERABLE status or close the associated block valves and remove power from the block valves; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If the block valves have been closed and power has been removed, restore at least one PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With three PORVs inoperable due to causes other than excessive leakage, within 1 hour either restore at least one PORV to OPERABLE status or close the associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV switch in the "close" position and remove power from its associated solenoid valve (do no enter action statement b for the resulting inoperable PORV); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

LIMITING CONDITION FOR OPERATION

f. With two block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place their associated PORV switches in the "close" position (do not enter action statement c for the resulting inoperable PORVs); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If the PORV switches have been placed in the "close" position, restore at least one block valve to OPERABLE status within 72 hours; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

g. With three block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place their associated PORV switches in the "close" position (do not enter action statement d for the resulting inoperable PORVs). Restore at least one block valve to OPERABLE status within the next hour; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel during MODE 3 or MODE 4 when the temperature of all RCS cold legs is greater than 300F with the block valve closed.

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 in order to meet the requirements of ACTION a., b., c., or d. in Specification 3.4.4.

4.4.4.3. The emergency power supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive power from the normal (air) supply to the emergency (nitrogen) supply.

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

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

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum 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 4.4.5.4.a.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. In addition to the 3% sample, all F* tubes will be inspected.

d. 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 following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

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

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

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

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

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

    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 or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve 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 or sleeve;

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

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

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

6) Repair Limit means the imperfection depth at or beyond which the tube or sleeve shall be removed from service by plugging or repaired by sleeving and is equal to 40% of the nominal tube or sleeve wall thickness. This definition does not apply to the area of the tubesheet region below the F* distance provided the tube is not degraded (i.e., no indications of cracking) within the F* distance. If a tube is sleeved due to degradation in the F* distance, then any defects in the tube below the sleeve will remain in service without repair.

The Babcock & Wilcox process (or method) equivalent to the method described in Topical Report BAW-2045(P)-A will be used.

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

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

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 after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) F* Distance is the distance into the tubesheet from the top face of the tubesheet or the top of the last hardroll, whichever is lower (further into the tubesheet) that has been conservatively chosen to be 2 inches.

11) F* TUBE is a tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of cracking) in the F* distance.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the repair 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 or repaired.

c. The results of inspections of F* tubes shall be reported to the Commission in a report, prior to the restart of the unit following the inspection. This report shall include:

1) Identification of F* tubes, and

2) Location and size of the degradation.

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 NOTATION:

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 deflective 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.	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.	N/A	N/A

S = 3 (N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generator 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. Either the Containment Floor and Equipment Sump Level System or the Flow Monitoring System, and

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

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

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

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

REACTOR COOLANT SYSTEM

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 primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

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

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

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

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

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 sump inventory or discharge at least once per 12 hours;

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

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

e. Monitoring the Reactor Head Flange Leakoff 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, and

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
NI60	Accumulator Discharge
NI71	Accumulator Discharge
NI59	Accumulator Discharge
NI70	Accumulator Discharge
NI82	Accumulator Discharge
NI94	Accumulator Discharge
NI81	Accumulator Discharge
NI93	Accumulator Discharge
NI134	Safety Injection (Hot Leg)
NI159	Safety Injection (Hot Leg)
NI156	Safety Injection (Hot Leg)
NI128	Safety Injection (Hot Leg)
NI124	Safety Injection (Hot Leg)
NI160	Safety Injection (Hot Leg)
NI157	Safety Injection (Hot Leg)
NI126	Safety Injection (Hot Leg)
NI129	Safety Injection (Hot Leg)
NI125	Safety Injection (Hot Leg)
NI165	Safety Injection/Residual Heat Removal (Cold Leg)
NI167	Safety Injection/Residual Heat Removal (Cold Leg)
NI169	Safety Injection/Residual Heat Removal (Cold Leg)
NI171	Safety Injection/Residual Heat Removal (Cold Leg)
NI175	Safety Injection/Residual Heat Removal (Cold Leg)
NI176	Safety Injection/Residual Heat Removal (Cold Leg)
NI180	Safety Injection/Residual Heat Removal (Cold Leg)
NI181	Safety Injection/Residual Heat Removal (Cold Leg)
ND1B*	Residual Heat Removal
ND2A*	Residual Heat Removal

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

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

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

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

At All Other Times:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

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

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*Limit not applicable with T_{avg} less than or equal to 250°F.

**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 1.0 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross specific activity.

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 1.0 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours;

b. With the specific activity of the 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; and

c. The provisions of Specification 3.0.4 are not applicable.

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

With the specific activity of the reactor coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram of gross specific activity, 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.

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.

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

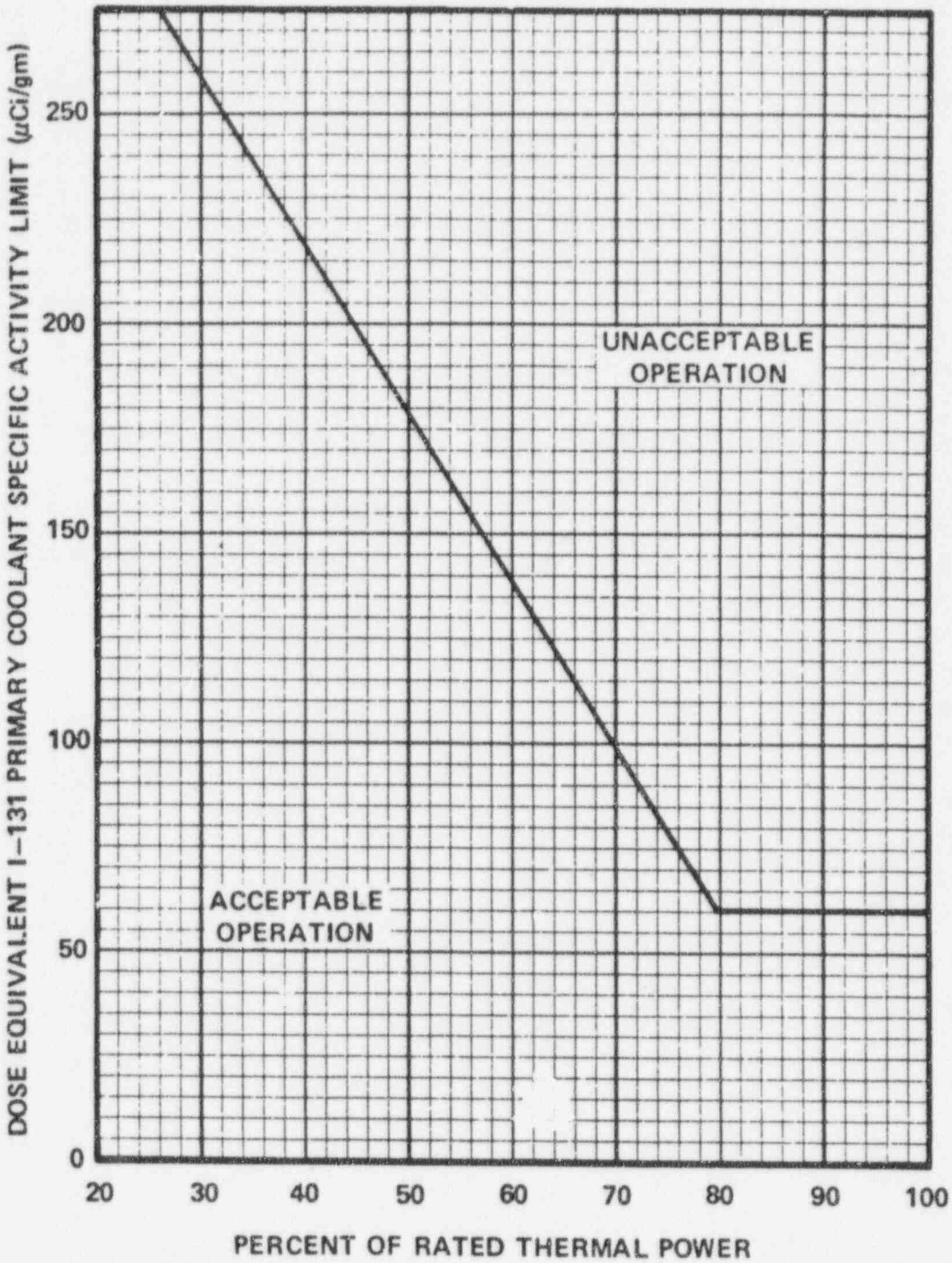


FIGURE 3.4-1
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

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 Specific Activity Determination**	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination***	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram 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 NOTATION

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

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

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the beta-gamma activity in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Maximum heatup rates as specified in Figure 3.4-2

b. Maximum cooldown rates as specified in Figure 3.4-3

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

APPLICABILITY: At all times.

ACTION:

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

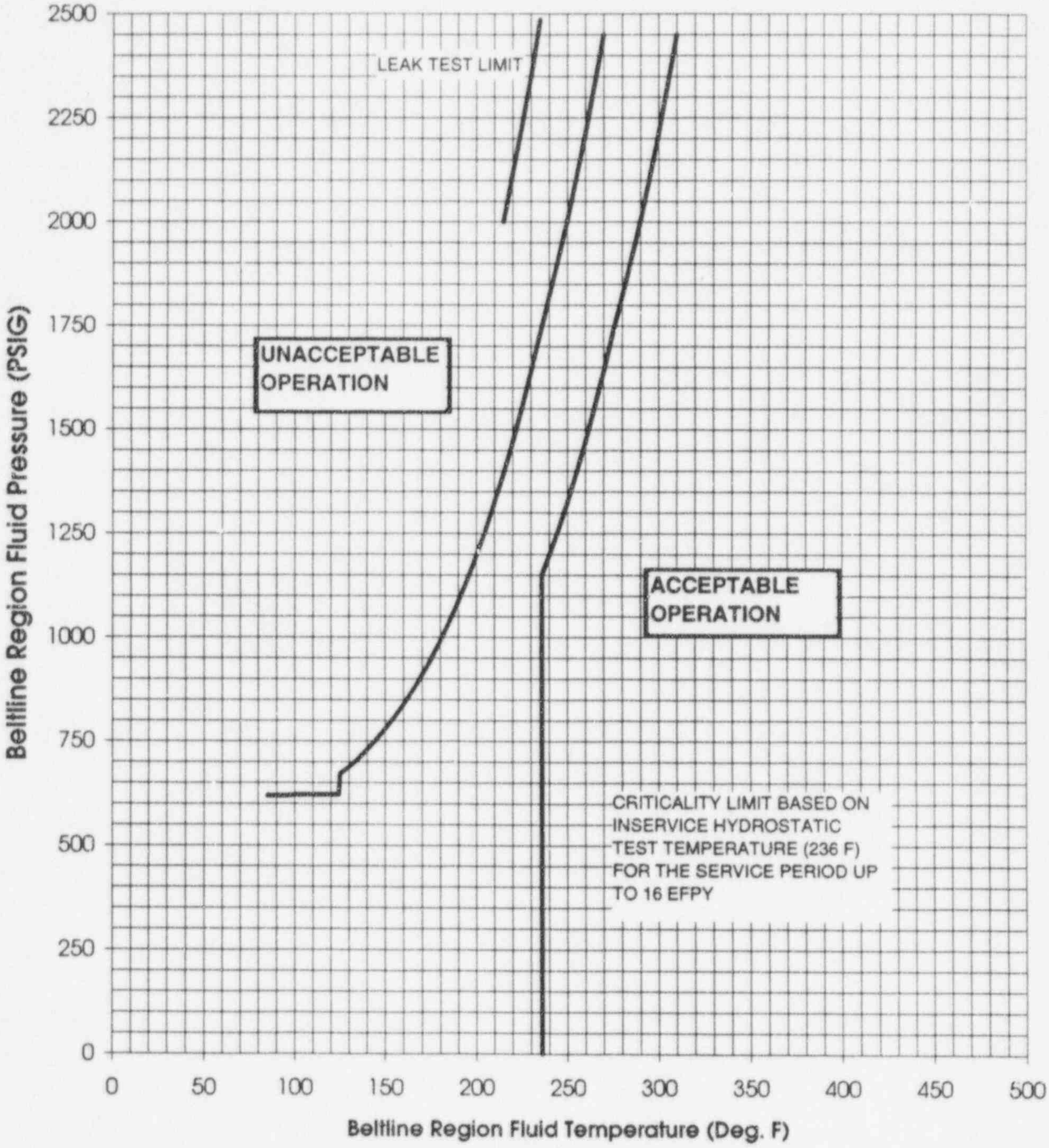
SURVEILLANCE REQUIREMENT

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 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

LIMITING MATERIALS: LOWER SHELL FORGING 04
LIMITING ART AT 16 EFPY:

1/4-t, 104 deg F
3/4-t, 73 deg F



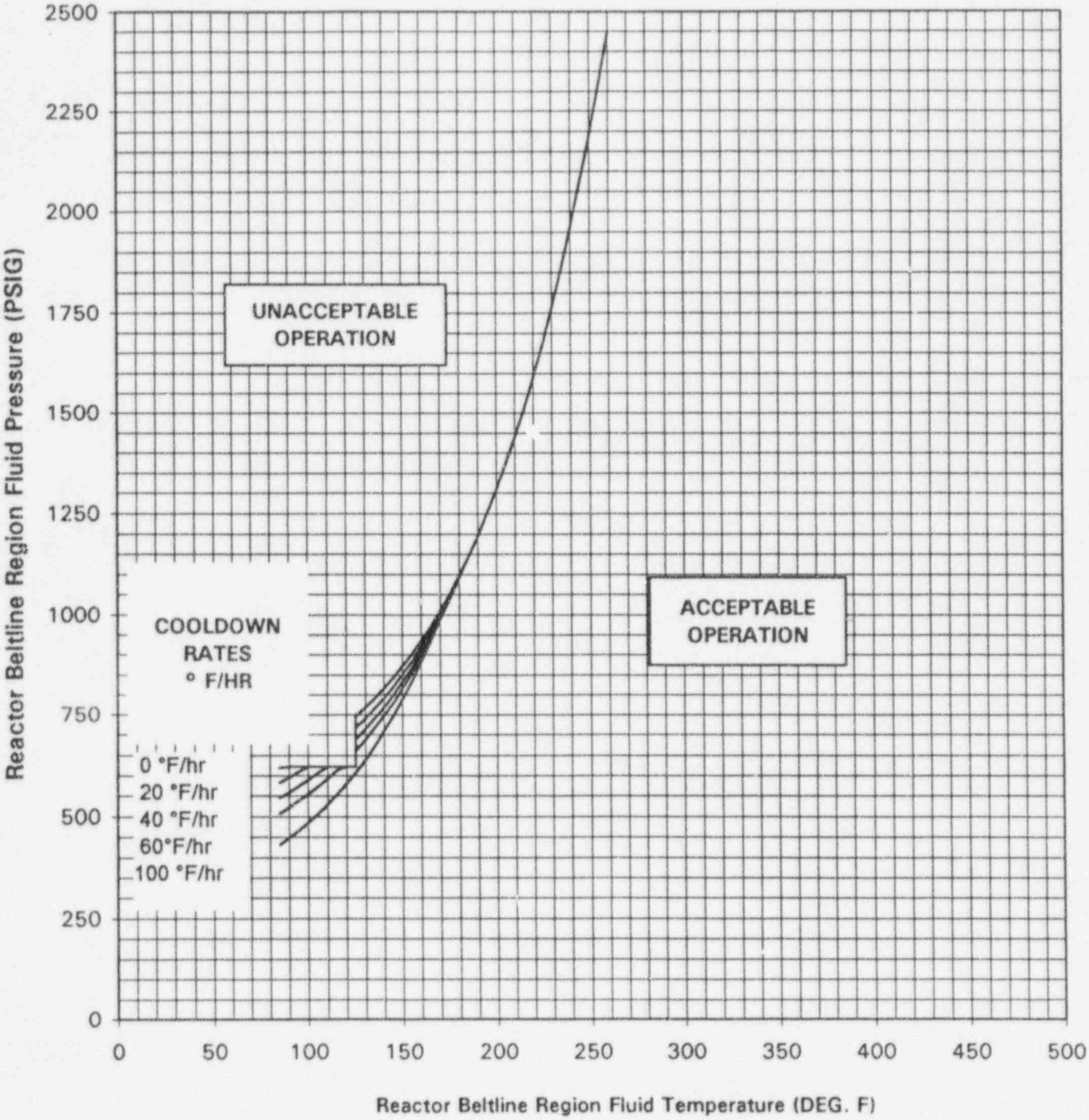
Reactor Coolant System Heatup Limitations
(Without Margins for Instrumentation Errors)
NRC REG GUIDE 1.99 Rev. 2
Applicable for the First 16 EFPY

Figure 3.4-2

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: LOWER SHELL FORGING 04

LIMITING ART AT 16 EFPY:
1/4-t, 104 deg F
3/4-t, 73 deg F



RCS Cooldown Limitations,
Cooldown Rates up to 100 deg F/HR
(without Margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the First 16 EFPY

Figure 3.4-3

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 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITION FOR OPERATION

3.4.9.3 As a minimum, a Low Temperature Overpressure Protection (LTOP) System shall be OPERABLE as follows:

a. A maximum of one Centrifugal Charging (NV) pump or one Safety Injection (NI) pump capable of injecting into the Reactor Coolant System (RCS) with all remaining NV and NI pump motor circuit breakers open or the discharge of the remaining NV and NI pumps isolated from the RCS by at least 2 valves with power removed#

AND

b. All accumulators isolated

AND

c. One of the following conditions met:

1. Two PORVs with a lift setting of ≤ 385 psig

OR

2. The RCS depressurized with a vent of ≥ 2.75 square inches.

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

ACTION:

a. With two or more Charging (NV) or Safety Injection(NI) pumps capable of injecting into the RCS*, immediately initiate action to restore a maximum of one NI or one NV pump capable of injecting into the RCS.

\# Two Charging pumps (NV or NI) maybe capable of injecting into the RCS during pump swap operation for ≤ 15 Minutes.

* One Safety Injection pump and one Charging pump, or two Charging pumps may be operated concurrently provided:

1. RHR suction relief valve (ND-3) is OPERABLE, and the RHR suction isolation valves (ND-1 and ND-2) are open and one of the following conditions is met:

    a. RCS cold leg temperature is greater than 167° F or

    b. RCS cold leg temperature is greater than 107° F and cooldown rate is less than 20° F per hour.

OR

2. Two PORVs secured in the open position with their associated block valves open and power removed.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (continued)

b. With an accumulator not isolated, isolate the affected accumulator within 1 hour. If required action is not met, either:

   1. Depressurize the accumulator to less than the maximum RCS pressure for the existing cold leg per Specification 3/4.4.9 within 12 hours,

   OR

   2. Increase RCS cold leg temperature to greater than or equal to 300° F within 12 hours.

c. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days. If required action is not met, depressurize the RCS and vent through at least a 2.75 square inch vent within 8 hours.

d. With one PORV inoperable in MODES 5 or 6, suspend all operations which could lead to a water-solid pressurizer. Restore the inoperable PORV to OPERABLE status within 24 hours. If required action is not met, either:

   1. Ensure RCS temperature is greater than 167° F, and ND-3 is OPERABLE, and ND-1 and ND-2 are open within one hour.

   OR

   2. Depressurize the RCS and vent through at least a 2.75 square inch vent within 8 hours.

e. With the LTOP system inoperable for any reason other than a., b., c., or d. above, depressurize the RCS and vent through at least a 2.75 square inch vent within 8 hours.

f. In the event that either the PORVs or the RCS vent 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 circumstance initiating the transient, the effect of the PORVs or vent on the transient, and any corrective action necessary to prevent recurrence.

g. 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, at least once per 31 days;

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 Once every 12 hours*, verify that an RCS vent of $\geq$ 2.75 square inches is open when the vent is used for overpressure protection.

4.4.9.3.3 Once every 12 hours, verify that each accumulator is isolated and that only one NV or NI pump is capable of injecting into the RCS.

4.4.9.3.4 Once every 12 hours, verify that RHR suction isolation valves ND-1 and ND-2 are open when RHR suction relief valve ND-3 is being used for overpressure protection.

4.4.9.3.5 Once every 72 hours, verify that the PORV block valve is open for each required PORV.

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* 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.

A PORV secured in the open position may be used to meet this vent requirement provided that its associated block valve is open and power is removed.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: All MODES.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.4.11 REACTOR VESSEL HEAD VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.11 Two reactor vessel head vent paths, each consisting of two valves in series powered from emergency buses shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With one of the above reactor vessel head paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both of the above reactor vessel head vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least two of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

2. Verifying flow through the reactor vessel head vent paths during venting during COLD SHUTDOWN or REFUELING.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6870 and 7342 gallons,

c. A boron concentration between the LCO limits presented in the Core Operating Limits Report,

d. A nitrogen cover-pressure of between 585 and 639 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than the lower LCO limit presented in the Core Operating Limits Report, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System 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 Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

c. With one accumulator inoperable due to boron concentration less than the lower LCO limit presented in the Core Operating Limits Report and:

   1) The volume weighted average boron concentration of the accumulators equal to the lower LCO limit presented in the Core Operating Limits Report or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

   2) The volume weighted average boron concentration of the accumulators less than the lower LCO limit presented in the Core Operating Limits Report but greater than the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report, restore the inoperable

---

*Reactor Coolant System pressure above 1000 psig.

LIMITING CONDITION FOR OPERATION (Continued)

accumulator to OPERABLE status or return the volume weighted average boron concentration of the accumulators to greater than the lower LCO limit presented in the Core Operating Limits Report and enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

3) The volume weighted average boron concentration of the accumulators equal to the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report or less, return the volume weighted average boron concentration of the accumulators to greater than the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2) Verifying that each cold leg injection accumulator isolation valve is open.

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 not resulting from normal makeup by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that power to the isolation valve operator is disconnected; and

d. At least once per 18 months by verifying proper operation of the power disconnect circuit.

4.5.1.1.2 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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
NI162A	Cold Leg Recirc.	Open*
NI121A	Hot Leg Recirc.	Closed
NI152B	Hot Leg Recirc.	Closed
NI183B	Hot Leg Recirc.	Closed
NI173A	RHR Pump Discharge	Open*
NI178B	RHR Pump Discharge	Open*
NI100B	SI Pump RWST Suction	Open
FW27A	RHR/RWST Suction	Open*
NI147A	SI Pump Mini flow	Open

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, unless the pumps and associated piping are in service or have been in service within 31 days, 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 interlock action of the RHR System from the Reactor Coolant System by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened.

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*Valves may be realigned to place RHR System in service and for testing pursuant to Specification 4.4.6.2.2.

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

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

1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection actuation and automatic switchover to Containment Sump Recirculation test signals, 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.

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

1)	Centrifugal charging pump	$\geq$ 2347 psid,
2)	Safety Injection pump	$\geq$ 1418 psid, and
3)	RHR pump	$\geq$ 166 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.

Boron Injection Throttle Valves	Safety Injection Throttle Valves
Valve Number	Valve Number
NI-480	NI-488
NI-481	NI-489
NI-482	NI-490
NI-483	NI-491

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 320 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 423 gpm, and

      b) The total pump flow rate is less than or equal to 675 gpm.

   3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 4025 gpm.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the RHR 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 capable of injecting into the RCS whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. Two charging pumps may be operable and operating for ≤ 15 minutes to allow swapping charging pumps. Additional requirements are provided by Specification 3.4.9.3

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 centrifugal charging pumps and Safety Injection pumps, not capable of injecting into the RCS shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each pump has been isolated from the RCS by at least two isolation valves with power removed from the valve operators at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

3/4.5.4 (Deleted)

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A contained borated water volume of at least 372,100 gallons,

b. A boron concentration between the limits presented in the Core Operating Limits Report,

c. A minimum solution temperature of 70°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With the 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

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2) Verifying the boron concentration of the water.

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

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves or operator action during periods when containment isolation valves are open under administrative control,** and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions;

b. By verifying that each containment air lock is in compliance with 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 P_a, 14.8 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.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and the annulus 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.

**The following valves may be opened on an intermittent basis under administrative control: NC-141, NC-142, WE-13, WE-23, VX-34, VX-40, FW-11, FW-13, FW-4.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.30% by weight of the containment air per 24 hours at P_a, 14.8 psig, or

2) Less than or equal to L_t, 0.14% by weight of the containment air per 24 hours at a reduced pressure of P_t, 7.4 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 bypass leakage rate of less than 0.07 L_a for all penetrations identified as secondary containment bypass leakage paths when pressurized to P_a.

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

ACTION:

With (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) the combined bypass leakage rate exceeding 0.07 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.07 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972 or the mass-plot method:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a, 14.8 psig, or at Pt, 7.4 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 either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1) Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a, or 0.25 L_t;

   2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at P_a, 14.8 psig, or P_t, 7.4 psig.

d. Type B and C tests shall be conducted with gas at P_a, 14.8 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks,

   2) Dual-ply bellows assemblies on containment penetrations between the containment building and the annulus, and

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

   4) Type C tests performed on containment penetrations M372, M373 without draining the glycol-water mixture from the seats of their diaphragm valves (NF-228A, NF-233B, and NF-234A), if meeting a zero indicated leakage rate (not including instrument error) for the diaphragm valves. These tests may be used in lieu of tests which are otherwise required by Section III.C.2(a) of 10 CFR 50, Appendix J to use air or nitrogen as

the test medium. The above required test pressure (P_a) and test interval are not changed by this exception.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.9.3 or 4.6.1.9.4, as applicable;

f. The combined bypass leakage rate shall be determined to be less than 0.07 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 14.8 psig, or P_t, 7.4 psig, during each Type A test;

g. Air locks shall be tested and demonstrated OPERABLE per Specification 4.6.1.3;

h. The space between each dual-ply bellows assembly on containment penetrations between the containment building and the annulus shall be vented to the annulus during Type A tests. Following completion of each Type A test, the space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3-5 psig to verify no detectable leakage or the dual-ply bellows assembly shall be subjected to a leak test with the pressure on the containment side of the dual-ply bellows assembly at P_a, 14.8 psig, or P_t, 7.4 psig, to verify the leakage to be within the limits of Specification 4.6.1.2f.;

i. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced Integrated Leakage Measurement System; and

j. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than 0.05 L_a at P_a, 14.8 psig.

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

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,

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

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of 14.8 psig,

b. By conducting overall air lock leakage tests at not less than P_a, 14.8 psig, and verifying the overall air lock leakage rate is within its limit:

   1) At lea. . nce 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, and

d. At least once per 6 months by conducting a pressure test to verify door seal integrity, with a measured leak rate of less than 15 standard cubic centimeters per minute.

#The provisions of Specification 4.0.2 are not applicable.

*This constitutes an exemption to Appendix J of 10 CFR 50.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained:

a. Between 75* and 100°F in the containment upper compartment, and

b. Between 100* and 120°F*** in the containment lower compartment.

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

ACTION:

With the containment average air temperature not conforming to the above limits, restore the air temperature to within the limits within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The primary containment upper compartment average air temperature shall be the weighted average** of ambient air temperature monitoring stations located in the upper compartment. Temperature readings will be obtained at least once per 24 hours from the elevation of 826 feet at the inlet of each upper containment ventilation unit.

4.6.1.5.2 The primary containment lower compartment average air temperature shall be the weighted average** of ambient air temperature monitoring stations located in the lower compartment. Temperature readings will be obtained at least once per 24 hours from the elevation of 745 feet at the inlet of each lower containment ventilation unit.

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*Lower limit may be reduced to 60°F in MODES 2, 3, and 4.

**The weighted average is the sum of each temperature multiplied by its respective containment volume fraction. In the event of inoperative temperature sensor(s), the weighted average shall be taken as the reduced total divided by one minus the volume fraction represented by the sensor(s) out of service.

***Containment lower compartment temperature may be between 120 and 125°F for up to 90 cumulative days per calendar year provided the lower compartment temperature average over the previous 365 days is less than 120°F.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73.

REACTOR BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the reactor building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

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

ACTION:

With the structural integrity of the reactor building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the reactor building shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the reactor building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the reactor building detected during the above required inspections shall be reported to the Commission pursuant to 10 CFR Sections 50.72, and 50.73.

ANNULUS VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 Two independent Annulus Ventilation Systems shall be OPERABLE.

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

ACTION:

a. With one Annulus Ventilation System inoperable for reasons other than the pre-heaters tested in 4.6.1.8.a and 4.6.1.8.d.5, 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.

b. With the pre-heaters tested in 4.6.1.8.a and 4.6.1.8.d.5 inoperable, restore the inoperable pre-heaters to OPERABLE status within 7 days, or file a Special Report in accordance with Specification 6.9.2 within 30 days specifying the reason for inoperability and the planned actions to return the pre-heaters to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.6.1.8 Each Annulus Ventilation System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS, by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the pre-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 ventilation system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance of 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 8000 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, and tested per ASTM D3803-89 has a methyl iodide penetration of less than 4%; and

   3) Verifying a system flow rate of 8000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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, and tested per ASTM D3803-89, has a methyl iodide penetration of less than 4%;

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 inches Water Gauge while operating the system at a flow rate of 8000 cfm ± 10%;

   2) Verifying that the system starts automatically on any Phase B Isolation test signal;

   3) Verifying that the filter cooling electric motor-operated bypass valves can be opened;

   4) Verifying that each system produces a negative pressure of greater than or equal to 0.5 inch W.G. in the annulus within 22 seconds after a start signal and that this negative pressure goes to -3.5 inches W.G. within 48 seconds after the start signal. Verifying that upon reaching a negative pressure of -3.5 inches W.G. in the annulus, the system switches into its recirculation mode of operation and that the time required for the annulus pressure to increase to -0.5 inch W.G. is greater than or equal to 278 seconds;

   5) Verifying that the pre-heaters dissipate 43.0 ± 6.4 kW at a nominal voltage of 600 VAC when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for DOP test aerosol while operating the system at a flow rate of 8000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 8000 cfm ± 10%.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 Each containment purge supply and/or exhaust isolation valve shall be OPERABLE and:

a. Each containment purge supply and/or exhaust isolation valve for the lower compartment (24-inch) and instrument room (12-inch and 24-inch) shall be sealed closed, and

b. The containment purge supply and/or exhaust isolation valve(s) for the upper compartment (24-inch) may be opened for up to 250 hours during a calendar year provided no more than one pair (one supply and one exhaust) are open at one time.

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

ACTION:

a. With any containment purge supply and/or exhaust isolation valve for the lower compartment or instrument room open or not sealed closed, close and/or seal closed that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment purge supply and/or exhaust isolation valve(s) for the upper compartment open for more than 250 hours during a calendar year, close any open 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.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.9.3 and/or 4.6.1.9.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 Each containment purge supply and/or exhaust isolation valve(s) for the lower compartment and instrument room shall be verified to be sealed closed at least once per 31 days.

4.6.1.9.2 The cumulative time that all containment purge supply and/or exhaust isolation valves for the upper compartment have been open during a calendar year shall be determined at least once per 7 days.

4.6.1.9.3 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed containment purge supply and/or exhaust isolation valve for the lower compartment and instrument room with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.

4.6.1.9.4 At least once per 3 months each containment purge supply and/or exhaust isolation valve for the upper compartment with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.01 L_a when pressurized to P_a.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2 Each Containment Spray System shall be demonstrated OPERABLE:

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

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 185 psig when tested pursuant to Specification 4.0.5;

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

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Phase B Isolation test signal,

   2) Verifying that each spray pump starts automatically on a Containment Phase B Isolation test signal,

   3) Verifying that the Containment Pressure Control System functions within the setpoint limits specified in Table 3.3-4, Item 6.

d. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE with isolation times less than or equal to the required isolation times.

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

ACTION:

With one or more containment isolation valve(s) 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.

e. The provisions of Specification 3.0.4 are not applicable provided that the affected penetration is isolated in accordance with ACTION b. or c. above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A Containment Isolation test signal, each Phase A isolation valve actuates to its isolation position,

b. Verifying that on a Phase B Containment Isolation test signal, each Phase B isolation valve actuates to its isolation position, and

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

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

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:

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.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using hydrogen gas mixtures to obtain calibration points of:

a. Zero volume percent hydrogen, and

b. Nine volume percent hydrogen.

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent Containment Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a Recombiner System functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW, and

b. At least once per 18 months by:

   1) Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

   2) Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

   3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

HYDROGEN CONTROL DISTRIBUTED IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 Both trains of the Hydrogen Mitigation System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one train of the Hydrogen Mitigation System inoperable, restore the inoperable system to OPERABLE status within 7 days or decrease the surveillance interval of Specification 4.6.4.3.a from 92 days to 7 days on the OPERABLE train until the inoperable train is returned to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each train of the Hydrogen Mitigation System shall be demonstrated OPERABLE:

a. At least once per 92 days by energizing the supply breakers and verifying that at least 32 of 33 igniters are energized,* and

b. At least once per 18 months by verifying the temperature of each igniter is a minimum of 1700°F.

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*Inoperable igniters must not be on corresponding redundant circuits which provide coverage for the same region.

3/4.6.5 ICE CONDENSER

ICE BED

LIMITING CONDITION FOR OPERATION

3.6.5.1 The ice bed shall be OPERABLE with:

a. The stored ice having a boron concentration of at least 1800 ppm boron as sodium tetraborate and a pH of 9.0 to 9.5,

b. Flow channels through the ice condenser,

c. A maximum ice bed temperature of less than or equal to 27°F,

d. A total ice weight of at least 2,099,790 pounds at a 95% level of confidence, and

e. 1944 ice baskets.

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

ACTION:

With the ice bed inoperable, restore the ice bed to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 The ice condenser shall be determined OPERABLE:

a. At least once per 12 hours by using the Ice Bed Temperature Monitoring System to verify that the maximum ice bed temperature is less than or equal to 27°F,

b. At least once per 9 months by:

1) Chemical analyses which verify that at least nine representative samples of stored ice have a boron concentration of at least 1800 ppm as sodium tetraborate and a pH of 9.0 to 9.5 at 20°C;

2) Weighing a representative sample of at least 144 ice baskets and verifying that each basket contains at least 1081 lbs of ice. The representative sample shall include 6 baskets from each of the 24 ice condenser bays and shall be constituted of

SURVEILLANCE REQUIREMENTS (Continued)

1 basket each from Radial Rows 1, 2, 4, 6, 8, and 9 (or from the same row of an adjacent bay if a basket from a designated row cannot be obtained for weighing) within each bay. If any basket is found to contain less than 1081 pounds of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The minimum average weight of ice from the 20 additional baskets and the discrepant basket shall not be less than 1081 pounds/basket at a 95% level of confidence.

The ice condenser shall also be subdivided into 3 groups of baskets, as follows: Group 1 - Bays 1 through 8, Group 2 - Bays 9 through 16, and Group 3 - Bays 17 through 24. The minimum average ice weight of the sample baskets from Radial Rows 1, 2, 4, 6, 8, and 9 in each group shall not be less than 1081 pounds/basket at a 95% level of confidence.

The minimum total ice condenser ice weight at a 95% level of confidence shall be calculated using all ice basket weights determined during this weighing program and shall not be less than 2,099,790 pounds; and

3) Verifying, by a visual inspection of at least two flow passages per ice condenser bay, that the accumulation of frost or ice on flow passages between ice baskets, past lattice frames, through the intermediate and top deck floor grating, or past the lower inlet plenum support structures and turning vanes is restricted to a thickness of less than or equal to 0.38 inch. If one flow passage per bay is found to have an accumulation of frost or ice with a thickness of greater than or equal to 0.38 inch, a representative sample of 20 additional flow passages from the same bay shall be visually inspected. If these additional flow passages are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one restricted flow passage per bay is evidence of abnormal degradation of the ice condenser.

c. At least once per 40 months by lifting and visually inspecting the accessible portions of at least two ice baskets from each one-third of the ice condenser and verifying that the ice baskets are free of detrimental structural wear, cracks, corrosion, or other damage. The ice baskets shall be raised at least 12 feet for this inspection.

ICE BED TEMPERATURE MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.2 The Ice Bed Temperature Monitoring System shall be OPERABLE with at least two OPERABLE RTD channels in the ice bed at each of three basic elevations (10'6", 30'9" and 55' above the floor of the ice condenser) for each one-third of the ice condenser.

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

ACTION:

a. With the Ice Bed Temperature Monitoring System inoperable, POWER OPERATION may continue for up to 30 days provided:

   1. The ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed;

   2. The last recorded mean ice bed temperature was less than or equal to 20°F and steady; and

   3. The ice condenser cooling system is OPERABLE with at least:

      a) 21 OPERABLE air handling units,

      b) 2 OPERABLE glycol circulating pumps, and

      c) 3 OPERABLE refrigerant units;

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

b. With the Ice Bed Temperature Monitoring System inoperable and with the Ice Condenser Cooling System not satisfying the minimum components OPERABILITY requirements of ACTION a.3. above, POWER OPERATION may continue for up to 6 days provided the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed and the last recorded mean ice bed temperature was less than or equal to 15°F and steady; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 The Ice Bed Temperature Monitoring System shall be determined OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours.

ICE CONDENSER DOORS

LIMITING CONDITION FOR OPERATION

3.6.5.3 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be closed and OPERABLE.

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

ACTION:

a. With one or more ice condenser doors open or otherwise inoperable (but capable of opening automatically), POWER OPERATION may continue for up to 14 days provided the ice bed temperature is monitored at least once per 4 hours and the maximum ice bed temperature is maintained less than or equal to 27°F; otherwise, restore the doors to their closed positions or OPERABLE status (as applicable) within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more ice condenser doors inoperable (not capable of opening automatically), restore all doors to OPERABLE status within 1 hour or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Inlet Doors - Ice condenser inlet doors shall be:

a. Continuously monitored and determined closed by the inlet door position monitoring system, and

b. Demonstrated OPERABLE during shutdown at least once per 18 months by:

1) Verifying that the torque required to initially open each door is less than or equal to 675 inch pounds;

2) Verifying that each door is capable of opening automatically in that it is not impaired by ice, frost, debris, or other obstruction;

3) Testing each one of the doors and verifying that the torque required to open each door is less than 195 inch-pounds when the door is 40 degrees open. This torque is defined as the "door opening torque" and is equal to the nominal door torque plus a frictional torque component;

SURVEILLANCE REQUIREMENTS (Continued)

4) Testing each one of the doors and verifying that the torque required to keep each door from closing is greater than 78 inch-pounds when the door is 40 degrees open. This torque is defined as the "door closing torque" and is equal to the nominal door torque minus a frictional torque component; and

5) Calculation of the frictional torque of each door tested in accordance with 3) and 4), above. The calculated frictional torque shall be less than or equal to 40 inch-pounds.

4.6.5.3.2 Intermediate Deck Doors - Each ice condenser intermediate deck door shall be:

a. Verified closed and free of frost accumulation by a visual inspection at least once per 7 days, and

b. Demonstrated OPERABLE at least once per 3 months during the first year after the ice bed is fully loaded and at least once per 18 months thereafter by visually verifying no structural deterioration, by verifying free movement of the vent assemblies, and by ascertaining free movement when lifted with the applicable force shown below:

	Door	Lifting Force
1)	Adjacent to crane wall	Equal to or less than 37.4 lbs,
2)	Paired with door adjacent to crane wall	Equal to or less than 33.8 lbs,
3)	Adjacent to containment wall	Equal to or less than 31.8 lbs, and
4)	Paired with door adjacent to containment wall	Equal to or less than 31.0 lbs.

4.6.5.3.3 Top Deck Doors - Each ice condenser top deck door shall be determined closed and OPERABLE at least once per 92 days by visually verifying:

a. That the doors are in place, and

b. That no condensation, frost, or ice has formed on the doors or blankets which would restrict their lifting and opening if required.

INLET DOOR POSITION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 The Inlet Door Position Monitoring System shall be OPERABLE.

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

ACTION:

With the Inlet Door Position Monitoring System inoperable, POWER OPERATION may continue for up to 14 days, provided the Ice Bed Temperature Monitoring System is OPERABLE and the maximum ice bed temperature is less than or equal to 27°F when monitored at least once per 4 hours; otherwise, restore the Inlet Door Position Monitoring System to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 The Inlet Door Position Monitoring System shall be determined OPERABLE by:

a. Performing a CHANNEL CHECK at least once per 7 days and within 4 hours after receiving an "Ice Condenser Inlet Door Open" alarm on the control room annunciator portion of the system,

b. Performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 18 months, and

c. Verifying that the Monitoring System correctly indicates the status of each inlet door as the door is opened and reclosed during its testing per Specification 4.6.5.3.1.

DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

LIMITING CONDITION FOR OPERATION

3.6.5.5 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be OPERABLE and closed.

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

ACTION:

With a personnel access door or equipment hatch inoperable or open except for personnel transit entry, restore the door or hatch to OPERABLE status or to its closed position (as applicable) within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.5.1 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined closed by a visual inspection prior to increasing the Reactor Coolant System T_{avg} above 200°F and after each personnel transit entry when the Reactor Coolant System T_{avg} is above 200°F.

4.6.5.5.2 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined OPERABLE by visually inspecting the seals and sealing surfaces of these penetrations and verifying no detrimental misalignments, cracks or defects in the sealing surfaces, or apparent deterioration of the seal material:

a. Prior to final closure of the penetration each time it has been opened, and

b. At least once per 10 years for penetrations containing seals fabricated from resilient materials.

CONTAINMENT AIR RETURN AND HYDROGEN SKIMMER SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent Containment Air Return and Hydrogen Skimmer Systems shall be OPERABLE.

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

ACTION:

With one Containment Air Return and Hydrogen Skimmer System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6.1 Each Containment Air Return and Hydrogen Skimmer System shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by:

a. Verifying that the air return and hydrogen skimmer fans start automatically on a Containment Phase B Isolation (S_p) test signal after a 9 ± 1 minute delay and operate for at least 15 minutes;

b. Verifying that during air return fan operation with the air return fan damper closed and with the bypass dampers open, the fan motor current is less than or equal to 32 amps when the fan speed is 870 ± 30 rpm;

c. Verifying that with the hydrogen skimmer fan operating and the motor operated valve in its suction line closed, the fan motor current is less than or equal to 21.5 amps when the fan speed is 3599 ± 20 rpm;

d. Verifying that with the air return fan off, the motor operated damper in the air return fan discharge line to the containment's lower compartment opens automatically with a 10 ± 1 second delay after a Containment Phase B Isolation (S_p) test signal;

e. Verifying that with the air return fan operating, the check damper in the air return fan discharge line to the containment's lower compartment is open;

f. Verifying that the motor operated valve in the hydrogen skimmer suction line opens automatically and the hydrogen skimmer fans receive a start permissive signal; and

g. Verifying that with the fan off, the return air fan check damper is closed.

4.6.5.6.2 At least once per 18 months, each Containment Air Return and Hydrogen Skimmer System shall be demonstrated OPERABLE by verifying that the containment pressure control system functions within the setpoint limits specified in Table 3.3-4, Item 6.

FLOOR DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.7 The ice condenser floor drains shall be OPERABLE.

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

ACTION:

With the ice condenser floor drain inoperable, restore the floor drain to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.7 Each ice condenser floor drain shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

a. Verifying that valve gate opening is not impaired by ice, frost or debris,

b. Verifying that the valve seat is not damaged,

c. Verifying that the valve gate opens when a force of less than or equal to 66 lbs is applied, and

d. Verifying that the drain line from the ice condenser floor to the containment lower compartment is unrestricted.

REFUELING CANAL DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.8 The refueling canal drains shall be OPERABLE.

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

ACTION:

With a refueling canal drain inoperable, restore the drain to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.8 Each refueling canal drain shall be demonstrated OPERABLE.

a. Prior to increasing the Reactor Coolant System temperature above 200°F after each partial or complete filling of the canal with water by verifying that the valves in the drain lines are locked open and that the drain is not obstructed by debris, and

b. At least once per 92 days by verifying, through a visual inspection, that there is no debris that could obstruct the drain.

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

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

ACTION:

With the divider barrier seal inoperable, restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:

a. Removing two divider barrier seal test coupons and verifying that the physical properties of the test coupons are within the acceptable range of values shown in Table 3.6-3; and

b. Visually inspecting at least 95% of the seal's entire length and:

   1) Verifying that the seal and seal mounting bolts are properly installed, and

   2) Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

TABLE 3.6-3

DIVIDER BARRIER SEAL
ACCEPTABLE PHYSICAL PROPERTIES

MEMBRANE TYPE SEALS	TENSILE STRENGTH
MK 10	39.7 lbs
MK 11	39.7 lbs

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-3.

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 COLD SHUTDOWN within the following 30 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 COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

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	(**)
2	(**)
3	(**)

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					LIFT SETTING(± 3%)***	ORIFICE SIZE
	Loop A	Loop B	Loop C	Loop D		
1.	SV 20	SV 14	SV 8	SV 2	1170 psig	12.174 in^2
2.	SV 21	SV 15	SV 9	SV 3	1190 psig	12.174 in^2
3.	SV 22	SV 16	SV 10	SV 4	1205 psig	16.00 in^2
4.	SV 23	SV 17	SV 11	SV 5	1220 psig	16.00 in^2
5.	SV 24	SV 18	SV 12	SV 6	1225 psig	16.00 in^2

---

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

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

***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 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) 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;

2) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

3) Verifying that the isolation valves in the auxiliary feedwater suction line from the upper surge tanks are open with power to the valve operators removed.

*Not applicable with steam pressure less than 900 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days on a STAGGERED BASIS by:

   1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 450 gpm; and

   2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 900 gpm when the secondary steam supply pressure is greater than 900 psig.* The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

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 upon receipt of an Auxiliary Feedwater Actuation test signal,

   2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal, and

   3) Verifying that the valve in the suction line of each auxiliary feedwater pump from the Nuclear Service Water System automatically actuates to its full open position within less than or equal to 13 seconds on a Low Suction Pressure test signal.

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* This verification is not required to be performed until 24 hours after achieving greater than or equal to 900 psig in the secondary side of the steam generator.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.3 The specific activity of the Secondary Coolant System shall be less than or equal to 0.10 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 0.10 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Specific Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross specific activity determination indicates concentrations greater than 10% of the allowable limit for radioiodine. b) Once per 6 months, whenever the gross specific activity determination indicates concentrations below 10% of the allowable limit for radioiodine.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.4 Each main steam line isolation valve (MSLIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1 - With one MSLIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within 2 hours.

MODES 2 AND 3 - With one MSLIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided:

a. The isolation valve is maintained closed, and

b. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.7.1.4 Each MSLIV shall be demonstrated OPERABLE by verifying full closure within 8 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

   2) Each component cooling water pump starts automatically on a Safety Injection and Station Blackout test signal.

3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With the unit specific portion of only one nuclear service water loop per unit OPERABLE, restore both unit specific loops to OPERABLE status within 72 hours or place the affected unit at least in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With only one of the shared portions as defined by Figure 3/4 7-1 of the Unit 1 and Unit 2 nuclear service water loops OPERABLE, restore the shared portion of the loops to OPERABLE status within 72 hours or place both units in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

   2) Each nuclear service water pump starts automatically on a Safety Injection and Station Blackout test signal.

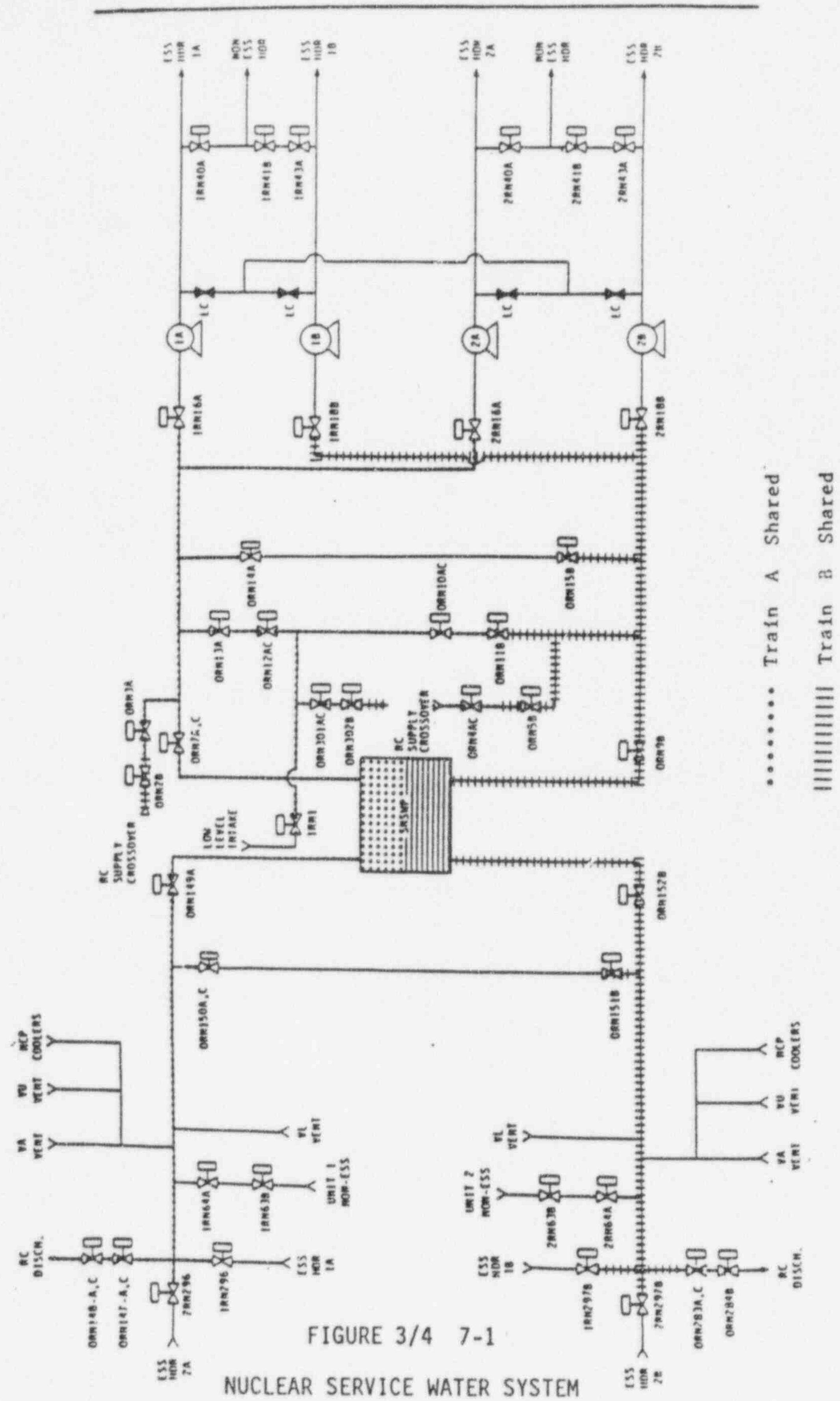

FIGURE 3/4 7-1

NUCLEAR SERVICE WATER SYSTEM

3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND

LIMITING CONDITION FOR OPERATION

3.7.5 The standby nuclear service water pond shall be OPERABLE with:

a. A minimum water level at or above elevation 739.5 feet Mean Sea Level, USGS datum, and

b. An average water temperature of less than or equal to 82°F at elevation 722 feet.

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

ACTION: (Units 1 and 2)

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The standby nuclear service water pond shall be determined OPERABLE:

a. At least once per 24 hours by verifying the water level to be within its limit,

b. At least once per 24 hours during the months of July, August and September by verifying the water temperature to be within its limit, and

c. At least once per 12 months by visually inspecting the dam and verifying no abnormal degradation, erosion, or excessive seepage.

3/4.7.6 CONTROL AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Area Ventilation Systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION: (Units 1 and 2)

MODES 1, 2, 3 and 4:

a. With one Control Area Ventilation System inoperable for reasons other than the heaters specified in 4.7.6.b and 4.7.6.e.4, 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.

b. With the heaters tested in 4.7.6.b and 4.7.6.e.4 inoperable, restore the inoperable heaters to OPERABLE status within 7 days, or file a Special Report in accordance with Specification 6.9.2 within 30 days, specifying the reason for inoperability and the planned actions to return the heaters to OPERABLE status.

MODES 5 and 6:

a. With one Control Area Ventilation System inoperable for reasons other than the heaters specified in 4.7.6.b and 4.7.6.e.4, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Area Ventilation System in the recirculation mode; and

b. With both Control Area Ventilation Systems inoperable for reasons other than the heaters specified in 4.7.6.b and 4.7.6.e.4, or with the OPERABLE Control Area Ventilation 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.

c. The provisions of Specification 3.0.4 are not applicable.

d. With the heaters tested in 4.7.6.b and 4.7.6.e.4 inoperable, restore the inoperable heaters to OPERABLE status within 7 days, or file a Special Report in accordance with Specification 6.9.2 within 30 days, specifying the reason for inoperability and the planned actions to return the heaters to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Area Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours, by verifying that the control room air temperature is less than or equal to 90°F;

b. At least once per 31 days on a STAGGERED TEST BASIS, by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters operating;

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 of 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 2000 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, and tested per ASTM D3803-89, has a methyl iodide penetration of less than 0.95%; and

   3) Verifying a system flow rate of 2000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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, and tested per ASTM D3803-89, has a methyl iodide penetration of less than 0.95%;

e. At least once per 18 months, by:

   1) Verifying that the pressure drop across the combined prefilters, HEPA filters and charcoal adsorber banks is less than 5 inches Water Gauge while operating the system at a flow rate of 2000 cfm ± 10%;

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that upon actuation of a diesel generator sequencer the system automatically switches into a mode of operation with flow through the HEPA filters and charcoal adsorber banks;

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch W.G. relative to the outside atmosphere during system operation; and

4) Verifying that the heaters dissipate 10 ± 1.0 kW at a nominal voltage of 600 VAC when tested in accordance with ANSI N510-1975.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 2000 cfm ± 10%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 2000 cfm ± 10%.

3/4.7.7 AUXILIARY BUILDING FILTERED VENTILATION EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 The Unit 1 and Unit 2 Auxiliary Building Filtered Ventilation Exhaust Systems shall be OPERABLE.

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

ACTION: (Units 1 and 2)

a. With one Auxiliary Building Filtered Ventilation Exhaust System filter package inoperable, restore the inoperable filter to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one Auxiliary Building Filtered Ventilation Exhaust System flowpath inoperable (except carbon and HEPA filter package components and except as addressed by c.1 and c.2 below) restore the inoperable flowpath 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.

c.1 With one Auxiliary Building Filtered Ventilation Exhaust System able to maintain a negative pressure but unable to maintain 0.125" W.G. at the ECCS pump room relative to outside atmosphere, restore system ability to maintain 0.125" W.G. within the next 7 days or be in a least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c.2 With one Auxiliary Building Filtered Ventilation Exhaust System unable to maintain a negative pressure at the ECCS pump room relative to outside atmosphere, restore system ability to maintain a negative pressure within the next 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With both Unit 1 and Unit 2 Auxiliary Building Filtered Ventilation Exhaust Systems inoperable, restore at least one inoperable system to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each unit's Auxiliary Building Filtered Ventilation Exhaust System filter package shall be demonstrated OPERABLE:

a. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or carbon adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

(1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance of 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 40,500 cfm ± 10% (both fans operating);

(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 an acceptance criteria for methyl iodide penetration of less than 10% at 30°C test temperature, and

b. After every 1440 hours of carbon adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets an acceptance criteria for methyl iodide penetration of less than 10% at 30°C test temperature, and

c. At least once per 18 months, by verifying that the pressure drop across the combined HEPA filters and carbon adsorber banks of less than 6 inches Water Gauge while operating the system at a flow rate of 40,500 cfm ± 10% (both fans operating), and

d. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 40,500 cfm ± 10% (both fans operating); and

e. After each complete or partial replacement of a carbon adsorber bank, by verifying that the carbon adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 40,500 cfm ± 10% (both fans operating).

4.7.7.2 Each Unit's Auxiliary Building Filtered Ventilation Exhaust System flowpath shall be demonstrated OPERABLE:

a. At least once per 31 days, by initiating, from the control room, flow through the HEPA filters and carbon adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or carbon adsorber housings, or (2) following

SURVEILLANCE REQUIREMENTS (Contir)

painting, fire, or chemical release in any ventilation zone communicating with the system, by verifying a system flow rate of 40,500 cfm ± 10% (both fans operating) during system operation when tested in accordance with ANSI N510-1980.

c. At least once per 18 months, by verifying that the system starts on a Safety Injection test signal and directs its exhaust flow through the HEPA filters and carbon adsorbers.

4.7.7.3 Each Unit's Auxiliary Building Filtered Ventilation Exhaust System shall be demonstrated OPERABLE, at least once per 18 months, by verifying that the system maintains the ECCS pump room at a negative pressure relative to outside atmosphere.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on non-safety-related systems and then only if the failure or the failure of the system on which they are installed would not have an adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4.
MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program 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 and may be treated independently. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each category of snubber shall be determined based upon the criteria provided in Table 4.7-2 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment No. 108.

TABLE 4.7-2

SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Notes 1 and 2)	Column A Extended Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. The categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category size and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as described by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

TABLE 4.7-2 (continued)

SNUBBER VISUAL INSPECTION INTERVAL

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

SURVEILLANCE REQUIREMENTS (Continued)

c. Refueling Outage Inspections

At each refueling, the systems which have the potential for a severe dynamic event, specifically, the main steam system (upstream of the main steam isolation valves) the main steam safety and power-operated relief valves and piping, auxiliary feedwater system, main steam supply to the auxiliary feedwater pump turbine, and the letdown and charging portion of the CVCS system shall be inspected to determine if there has been a severe dynamic event. In case of a severe dynamic event, mechanical snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the mechanical snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying freedom of motion using one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; (3) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced or repaired before returning to power. The requirements of Specification 4.7.8b. are independent of the requirements of this specification.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify: (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections "shall be classified as unacceptable and may be reclassified acceptable" 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 4.7.8f. A hydraulic snubber found with the fluid port uncovered and all hydraulic snubbers found connected to an inoperable common reservoir shall be classified as unacceptable and may be reclassified acceptable by functionally testing each snubber starting with the piston in the as-found setting, extending the piston rod in the tension direction.

e. Functional Tests

During the first refueling shutdown and at least once per refueling thereafter, a representative sample of snubbers shall be tested using one of the following sample plans. The large bore steam generator hydraulic snubbers shall be treated as a separate population for functional test purposes. A 10% random sample from previously untested snubbers shall be tested at least once per refueling outage until the entire population has been tested. This testing cycle shall then begin anew. For each large bore steam

e. Functional Tests (Continued)

generator hydraulic snubber that does not meet the functional test acceptance criteria, at least 10% of the remaining population of untested snubbers for that testing cycle shall be tested. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC shall be notified of the sample plan selected prior to the test period.

1) At least 10% of the snubbers required by Specification 3.7.8 shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.8f., an additional 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested; or

2) A representative sample of the snubbers required by Specification 3.7.8 shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers found not meeting the acceptance requirements of Specification 4.7.8f (failures). The cumulative number of snubbers tested is denoted by "N." Test results shall be plotted sequentially in the order of sample assignment (i.e., each snubber shall be plotted by its order in the random sample assignments, not by the order of testing). If at any time the point plotted falls in the "Accept" region, testing of snubbers may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers shall be tested until the point falls in the "Accept" region, or all the snubbers required by Specification 3.7.8 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, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of fifty-five (55) snubbers shall be functionally tested. For each snubber 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. This can be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber should be plotted as soon as it is tested. If the point plotted falls on or below the "Accept" line, testing may be discontinued. If the point plotted falls above the "Accept" line, testing must continue unless all snubbers have been tested.

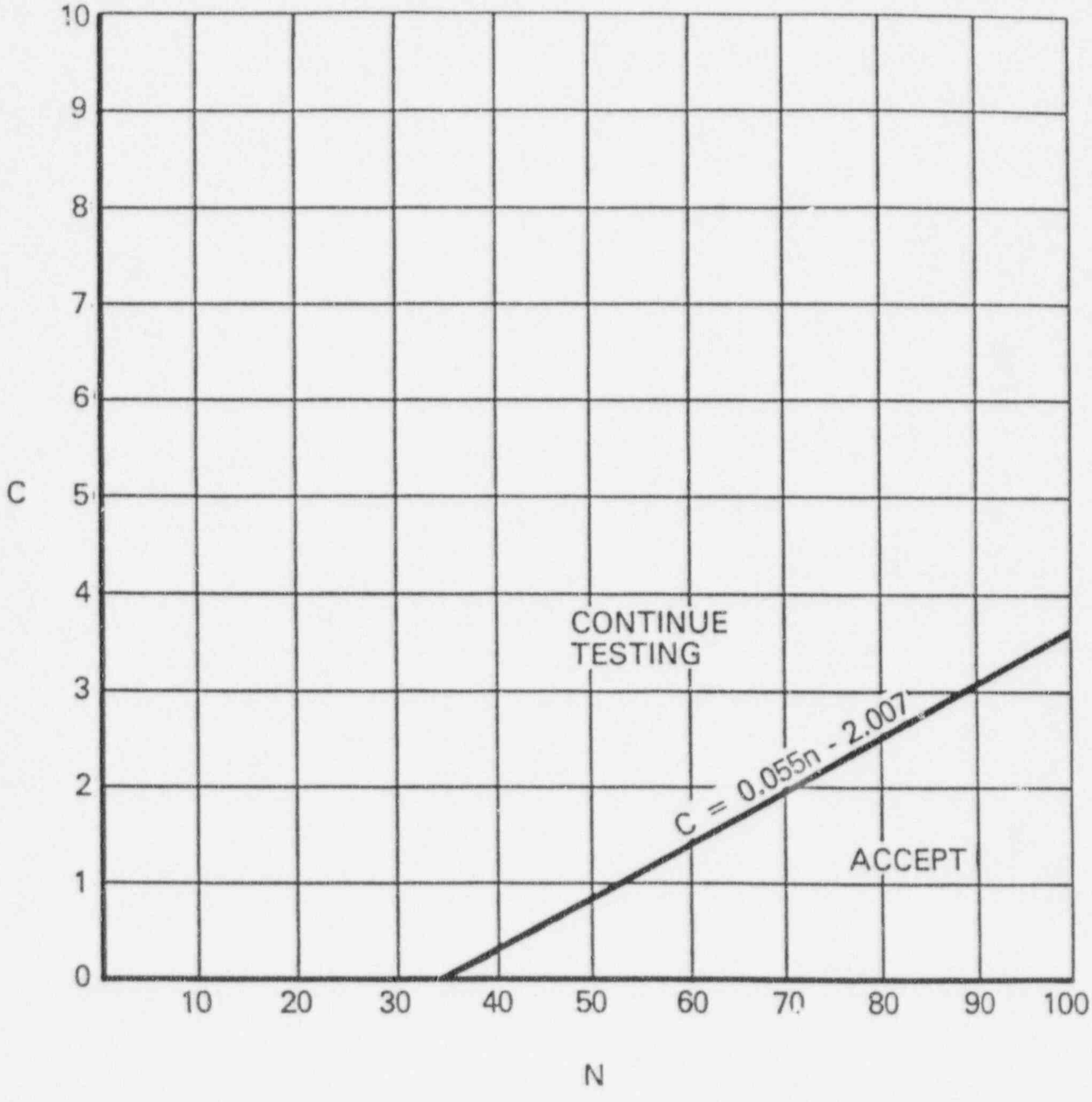

FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

e. Functional Tests (Continued)

The representative samples for the functional test sample plans shall be randomly selected from the snubbers required by Specification 3.7.8 and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of sizes, and capacities. Snubbers placed in the same locations 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 testing results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both direction 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 activate 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 evaluated in a manner to ensure their OPERABILITY. This testing requirement shall be independent of the requirements stated in Specification 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result 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 freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the expected service life will not be exceeded during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.3 Reports - A report shall be prepared 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 7.10 DELETED

3/4 7.11 DELETED

3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature of each area shown in Table 3.7-6 shall be maintained within the limits indicated in Table 3.7-6.

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. For more than 8 hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 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. Containment Spray Pump Rooms	145
2. Miscellaneous Terminal Cabinets	
a. TB1208-1209 (Turbine Bldg.)	150
b. TB1496 (Fuel Bldg.)	150
3. Residual Heat Removal Pump Rooms	145
4. Diesel Generator Rooms	125
5. Spent Fuel Pool Cooling Pump Room	145

3/4.7.13 GROUNDWATER LEVEL

LIMITING CONDITION FOR OPERATION

3.7.13 The groundwater level shall be maintained at elevations less than the values in Table 3.7-7 for the five (5) Auxiliary Building monitors listed in Table 3.7-7.

APPLICABILITY: At all times.

ACTION: For Units 1 and 2.

If groundwater level for any three (3) of the five (5) monitors is above the values shown in Table 3.7-7, take the following actions:

1. Within one hour, reduce the groundwater level to below the values shown in Table 3.7-7; or,

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

SURVEILLANCE REQUIREMENTS

4.7.13.1 During each shift, the groundwater level shall be demonstrated to be within the values of Table 3.7-7 by the absence of alarms or by visual observation of the monitor level gauge.

4.7.13.2 Each groundwater level monitor instrument/loop for locations listed in Table 3.7-7 shall be demonstrated OPERABLE at least once per year by the performance of a loop calibration and operational test.

TABLE 3.7-7

AUXILIARY BUILDING GROUNDWATER LEVEL MONITORS

LOCATION	INTERIOR/ EXTERIOR ELEVATION	UNIT
	(Feet - Mean Sea Level)	
PP-51	Interior 731' - 0"	1
QQ-56	Interior 731' - 0"	1 & 2
PP-61	Interior 731' - 0"	2
West Wall	Exterior 731' - 0"	1
East Wall	Exterior 731' - 0"	2

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the Onsite Essential Auxiliary Power System, and

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

   1) A separate day tank containing a minimum volume of 120 gallons of fuel,

   2) A separate Fuel Storage System containing a minimum volume of 39,500 gallons of fuel,

   3) A separate fuel transfer pump.

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

ACTION:

a. With an offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; separately demonstrate the operability of two diesel generators by performing Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5) within 24 hours unless this surveillance was performed within the previous 24 hours, or unless the diesel is operating, 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. source by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; demonstrate the operability of the remaining diesel generator by performing Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5) within 8 hours unless this surveillance was performed within the previous 24 hours, or unless the diesel is operating**; 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; with the

*A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5).

**This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

ACTION: (Continued)

diesel generator restored to OPERABLE status, follow action statement a; with the offsite circuit restored to OPERABLE status, follow action statement d.

c. With one diesel generator inoperable in addition to ACTION b. or d. 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

   2. When in MODE 1, 2, or 3 with a steam pressure greater than 900 psig, the steam-driven auxiliary feedwater pump is OPERABLE.

   If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With a diesel generator of the above required A.C. electrical power sources inoperable*, demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter; and unless the inoperability of the diesel was due to preplanned testing or maintenance, demonstrate the OPERABILITY of the remaining diesel generator by performing Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5) within 24 hours or unless the diesel is operating**, restore 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.

e. With two of the above required offsite A.C. circuits inoperable, separately demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5) within 8 hours, 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 and in COLD SHUTDOWN within the following 30 hours. With only one offsite source restored, follow action statement a.

f. With two of the above required diesel generators inoperable*, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once

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*A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5).

**This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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. With one diesel generator restored, follow action statement d.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Essential Auxiliary Power System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the day tank,

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least 488 rpm in less than or equal to 11 seconds.* The generator voltage and frequency shall be at least 4160 volts and 57 Hz within 11 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

---

*The diesel generator start (11 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded to greater than or equal to 3000 kW in less than or equal to 60 seconds, and to 4000 kW within 10 minutes and operates for at least 60 minutes, and

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By removing accumulated water:

1) From the day tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and

2) From the storage tank at least once per 31 days.

c. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the 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:

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 at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2) By verifying within 31 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.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks 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.

e. At least once per 18 months, by:

1) Subjecting the diesel to an inspection, during shutdown, in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;

2) Verifying, during shutdown, the generator capability to reject a load of greater than or equal to 576 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz;

3) Verifying, during shutdown, the generator capability to reject a load of 4000 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, during shutdown, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected blackout loads through the load sequencer and operates for greater than or equal to 5 minutes while the generator is loaded with the blackout loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ESF actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be at least 4160 volts and 57 Hz within 11 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within 4160 ± 420 volts and 60 ± 1.2 Hz during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

a) Verifying, during shutdown, deenergization of the emergency busses and load shedding from the emergency busses;

SURVEILLANCE REQUIREMENTS (Continued)

b) Verifying, during shutdown, the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

c) Verifying, during shutdown,* that all automatic diesel generator trips, except engine overspeed, lube oil pressure, generator time overcurrent, and generator differential are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) [Deleted, Left Blank]

8) Verifying, during shutdown, the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded between 4200 kW and 4400 kW** and during the remaining 22 hours of this test, the diesel generator shall be loaded between 3800 kW and 4000 kW.** The generator voltage and frequency shall be at least 4160 volts and 57 Hz within 11 seconds after the start signal. The steady-state generator voltage and frequency shall be maintained within 4160 ± 420 volts and 60 ± 1.2 Hz during this test. Within 5 minutes of shutting down the diesel generator, restart the diesel generator and verify that the generator voltage and frequency reaches at least 4160 volts and 57 Hz within 11 seconds.***

9) Verifying that the auto-connected loads to each diesel generator do not exceed the 2-hour rating of 4400 kW;

10) Verifying, during shutdown, the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
   b) Transfer its loads to the offsite power source, and
   c) Be restored to its standby status.

11) Verifying, during shutdown, that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

SURVEILLANCE REQUIREMENTS (Continued)

12) 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;

13) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block are within the tolerances shown in Table 4.8-2;

14) Verifying, during shutdown,* that the following diesel generator lockout features prevent diesel generator starting only when required:

a) Turning gear engaged, or
b) Emergency stop.

15) Verifying, during shutdown, that with all diesel generator air start receivers pressurized to less than or equal to 220 psig and the compressors secured, the diesel generator starts at least 2 times from ambient conditions and accelerates to at least 488 rpm in less than or equal to 11 seconds.

f. 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 488 rpm in less than or equal to 11 seconds; and

g. 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

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*This Surveillance Requirement may be performed in conjunction with periodic preplanned preventative maintenance activity that causes the diesel generator to be inoperable provided that performance of the surveillance requirement does not increase the time the diesel generator would be inoperable for the PM activity alone.

**Diesel generator loadings for the purpose of this surveillance may be in accordance with vendor recommendations. The purpose of the load range is to prevent overloading the engine and momentary excursions outside of the range shall not invalidate the test.

***If there is a test failure during the 24-hour test run, the hot restart test can be performed prior to completing the 24-hour test provided the diesel generator had operated for at least 2 hours loaded between 3800 and 4000 kW.**

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 non-valid, shall be reported to the Commission pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

4.8.1.1.4 Diesel Generator Batteries - Each diesel generator 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

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

   2) The overall battery voltage is greater than or equal to 125 volts under a float charge.

b. At least once per 18 months by verifying that:

   1) The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration;

   2) The battery-to-battery and terminal connections are clear, tight, free of corrosion and coated with anti-corrosion material; and

   3) The battery capacity is adequate to supply and maintain in OPERABLE status its emergency loads when subjected to a battery service test.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS *	NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
≤ 1	≤ 4	Once per 31 days
≥ 2**	≥ 5	Once per 7 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, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new conditions is completed, provided that the overhaul including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer <u>and</u> if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine surveillance requirements of Specifications 4.8.1.1.2.a.4) and 4.8.1.1.2.a.5); the remaining four tests in accordance with the 184-day requirements specified in the footnote to Specification 4.8.1.1.2.a.4) and Specification 4.8.1.1.2.a.5). If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed <u>and</u> the number of failures in the last 20 valid demands has been reduced to one.

TABLE 4.8-2

LOAD SEQUENCING TIMES

LOAD GROUP NUMBER	SEQUENCE TIME (Seconds)
Initiate Timer (T_o)	9.7 ± 0.3
1 (T_1)	T_o + 0.9 ± 0.1
2 (T_2)	T_o + 5.6 ± 0.4
3 (T_3)	T_o + 9.4 ± 0.6
4 (T_4)	T_o + 14.1 ± 0.9
5 (T_5)	T_o + 18.4 ± 1.2
6 (T_6)	T_o + 23.1 ± 1.4
7 (T_7)	T_o + 28.3 ± 1.7
8 (T_8)	T_o +530.0 ± 60.0
9 (T_9)	T_8 + 56.0 ± 4.0
10 (T_{10})	T_8 +112.3 ± 7.0

LIMITING CONDITION FOR OPERATION

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

a. One circuit between the offsite transmission network and the Onsite Essential Auxiliary Power System, and

b. One diesel generator with:

   1) A day tank containing a minimum volume of 120 gallons of fuel,

   2) A Fuel Storage System containing a minimum volume of 28,000 gallons of fuel, and

   3) A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 4.5 square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5), 4.8.1.1.3, and 4.8.1.1.4.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following D.C. channels shall be OPERABLE and energized:

a. Channel 1 consisting of 125-Volt D.C. Bus No. EVDA, 125-Volt D.C. Battery Bank No. EVCA and a full-capacity charger,*#

b. Channel 2 consisting of 125-Volt D.C. Bus No. EVDB, 125-Volt D.C. Battery Bank No. EVCB and a full-capacity charger,*#

c. Channel 3 consisting of 125-Volt D.C. Bus No. EVDC, 125-Volt D.C. Battery Bank No. EVCC and a full-capacity charger,*# and

d. Channel 4 consisting of 125-Volt D.C. Bus No. EVDD, 125-Volt D.C. Battery Bank No. EVCD and a full-capacity charger,*#

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

ACTION: (Units 1 and 2)

a. With one 125-volt D.C. bus inoperable or not energized, restore the inoperable bus to OPERABLE and energized status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125-volt D.C. battery and/or its normal and standby chargers inoperable or not energized, either:

   1. Restore the inoperable battery and/or charger to OPERABLE and energized 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, or

   2. Energize the associated bus with an OPERABLE battery bank via OPERABLE tie breakers within 2 hours; operation may then continue for up to 72 hours from time of initial loss of OPERABILITY, otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*A vital bus may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided the vital busses associated with the other battery banks are OPERABLE and energized.

#During periods of station modification associated with battery, main and tie breaker replacement only, the loads of a DC bus may be energized from a same train DC bus via temporary cables and breakers connecting to the same train DC bus directly and bypassing the de-energized DC bus. A one time allowable outage time up to 112 hours is granted for each DC bus, one at a time, to allow for replacement of these breakers. Footnote * shall not be applied to any of the busses during the 112 hour period.

SURVEILLANCE REQUIREMENTS

4.8.2.1.1 Each D.C. channel shall be determined OPERABLE and energized with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment, indicated power availability from the charger and battery, and voltage on the bus of greater than or equal to 125 volts.

4.8.2.1.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days:

   1) Verifying that the parameters in Table 4.8-3 meet the Category A limits, and

   2) Verifying total battery terminal voltage is greater than or equal to 125 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge (battery terminal voltage below 110 volts), or battery overcharge (battery terminal voltage above 150 volts), by:

   1) Verifying that the parameters in Table 4.8-3 meet the Category B limits,

   2) Verifying there is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

   3) Verifying that the average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

   1) The cells, cell plates (if visible), and battery racks show no visual indication of physical damage or abnormal deterioration,

   2) The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

   3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms; and

   4) The battery charger will supply at least 400 amperes at a minimum of 125 volts for at least 1 hour.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by verifying that the battery capacity is adequate to either:

   1) Supply and maintain in OPERABLE status all of the actual emergency loads for 1 hour when the battery is subjected to a battery service test, or

   2) Supply a dummy load of greater than or equal to 440 amperes for 60 minutes while maintaining the battery terminal voltage greater than or equal to 105 volts.

e. At least once per 60 months by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval, this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1.2d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 80% of the manufacturer's rating.

TABLE 4.8-3

BATTERY SURVEILLANCE REQUIREMENTS (Gould Cells)

	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 (c)	> 2.07 volts
Specific Gravity (a)	≥ 1.200 (b)	≥ 1.195 Average of all connected cells > 1.205	Not more than .020 below the average of all connected cells or ≥ 1.195 Average of all connected cells ≥ 1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps when on charge.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category 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.

TABLE 4.8-3 (Continued)

BATTERY SURVEILLANCE REQUIREMENTS (AT&T Cells)

	Category A (1)		Category B(2)	Category C(3)
Parameter	Limits for each designated pilot cell		Limits for each connected cell	Allowable value for each connected cell
Electrolyte Level	$\geq$ Minimum level indication mark, and $\leq$ 1/4" above maximum level indication mark		$\geq$ Minimum level indication mark, and $\leq$ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.20 Volts		$\geq$ 2.17 Volts (4)	> 2.14 Volts
Specific (5) Gravity	$\geq$ 1.285 (6)	CELL	$\geq$ 1.280	Not more than 0.020 below the average of all connected cells or $\geq$ 1.280
		BATTERY	Average of all connected cells > 1.285 (7)	Average of all connected cells $\geq$ 1.280 (6)(7)

(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 C measurements are taken and found to be within their allowable values. All Category B parameter(s) must be within limits in 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 C parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category C parameter not within its allowable value indicates an INOPERABLE battery.

(4) Corrected for average electrolyte temperature.

(5) Corrected for electrolyte temperature and level.

(6) Or battery charging current is less than 2 amps when on float charge.

(7) With no more than 5 cells at the minimum limits.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following D.C. electrical equipment and bus shall be OPERABLE and energized:

a. Two - 125-volt D.C. busses, and

b. Two - 125-volt D.C. battery banks and chargers associated with the above D.C. busses.

APPLICABILITY: MODES 5 and 6.

ACTION: (Units 1 and 2)

With less than the above complement of D.C. equipment and bus OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2.1 The above required 125-volt D.C. busses shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the bus.

4.8.2.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Specification 4.8.2.1.2.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following A.C. electrical busses and inverters shall be OPERABLE and energized with tie breakers open both between redundant busses within the unit and between the two units:

a. 4160-Volt Emergency Bus ETA,
b. 4160-Volt Emergency Bus ETB,
c. 600-Volt Emergency Bus ELXA,
d. 600-Volt Emergency Bus ELXB,
e. 600-Volt Emergency Bus ELXC,
f. 600-Volt Emergency Bus ELXD,
g. 120-Volt A.C. Vital Bus EKVA energized from Inverter EVIA connected to D.C. Channel 1,*#
h. 120-Volt A.C. Vital Bus EKVB energized from Inverter EVIB connected to D.C Channel 2,*#
i. 120-Volt A.C. Vital Bus EKVC energized from Inverter EVIC connected to D.C. Channel 3,*# and
j. 120-Volt A.C. Vital Bus EKVD energized from Inverter EVID connected to D.C. Channel 4.*#

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

ACTION:

a. With less than the above complement of A.C. busses OPERABLE and energized, restore the inoperable busses to OPERABLE and energized status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one inverter inoperable, energize the associated A.C. Vital Bus within 8 hours; restore the inoperable inverter to OPERABLE and energized status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified A.C. busses and inverters shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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*An inverter may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided: (1) its vital bus is OPERABLE and energized, and (2) the vital busses associated with the other battery banks are OPERABLE and energized. An inverter may be disconnected from its D.C. source for up to 72 hours provided the conditions of ACTION b. of Specification 3.8.2.1 are satisfied.

#During periods of station modification associated with battery, main and tie breaker replacement only, two channel related inverters maybe energized from a same train DC bus via temporary cables and breakers connecting to the same train DC bus directly and bypassing the associated de-energized DC bus. A one time allowable outage time up to 112 hours is granted for each DC bus, one at a time, to allow for replacement of these breakers. Footnote * shall not be applied to any of the busses during the 112 hour period.

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following A.C. electrical busses and inverters shall be OPERABLE and energized:

a. One - 4160-volt emergency bus,

b. Two - 600-volt emergency busses in a single train, and

*c. Two - 120-volt A.C. vital busses energized from their respective inverters connected to their respective D.C. channels.

APPLICABILITY MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours. Specification 3.8.3.2c. applies to both Units 1 and 2.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified A.C. busses and inverters shall be determined energized in the required manner at least once per 7 days by verifying correct alignment and indicated voltage on the busses.

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*Required for both Units 1 and 2.

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 shown in FSAR Chapter 16 shall be OPERABLE.

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

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) shown in FSAR Chapter 16 inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices shown in FSAR Chapter 16 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 as specified in FSAR Chapter 16, 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. For the lower voltage circuit breakers the nominal Trip Setpoint and overcurrent response times are listed in FSAR Chapter 16. 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; and

3) A fuse inspection and maintenance program will be maintained to ensure that:

1. The proper size and type of fuse is installed,
2. The fuse shows no sign of deterioration, and
3. The fuse connections are tight and clean.

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.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to the minimum boron concentration specified in the Core Operating Limits Report.

APPLICABILITY: MODE 6*, with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to the minimum boron concentration specified in the Core Operating Limits Report, 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 NV-250 shall be verified closed under administrative control at least once per 72 hours; or, NV-131, NV-140, NV-176, NV-468, NV-808, and either NV-132 or NV-1026 shall be verified closed under administrative control at least once per 12 hours when necessary to makeup to the RWST during refueling operations.

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*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE and operating with Alarm Setpoints at 0.5 decade above steady-state count rate, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 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 hatch closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Exhausting through OPERABLE Reactor Building Containment Purge Exhaust System HEPA filters and charcoal adsorbers.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment building penetrations shall be determined to be in its required condition 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.

4.9.4.2 The Reactor Building Containment Purge Exhaust System shall be demonstrated OPERABLE:

a. 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 system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance of 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 21,000 cfm ± 10% (both exhaust units operating);

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

3) Verifying a system flow rate of 21,000 cfm ± 10% (both exhaust units operating) during system operation when tested in accordance with ANSI N510-1975.

b. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

c. At least once per 18 months, by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 21,000 cfm ± 10% (both exhaust units operating);

d. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 21,000 cfm ± 10% (both exhaust units operating); and

e. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 21,000 cfm ± 10% (both exhaust units operating).

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1) A minimum capacity of 3250 pounds, and

   2) An overload cutoff limit less than or equal to 2900 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

   1) A minimum capacity of 610 pounds, and

   2) A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cutoff when the crane load exceeds 2900 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 610 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 3000 pounds* shall be prohibited from travel over fuel assemblies in the storage pool. Truck casks shall be carried along the path outlined in Figure 3.9-1 in the fuel pit and fuel pool area.

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 Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The weight of each load, other than a fuel assembly and control rod, shall be verified to be less than 3000 pounds prior to moving it over fuel assemblies.*

*Weir gates of the spent fuel pool may be moved by crane over the stored fuel provided the spent fuel has decayed for at least 17.5 days since last being part of a core at power.

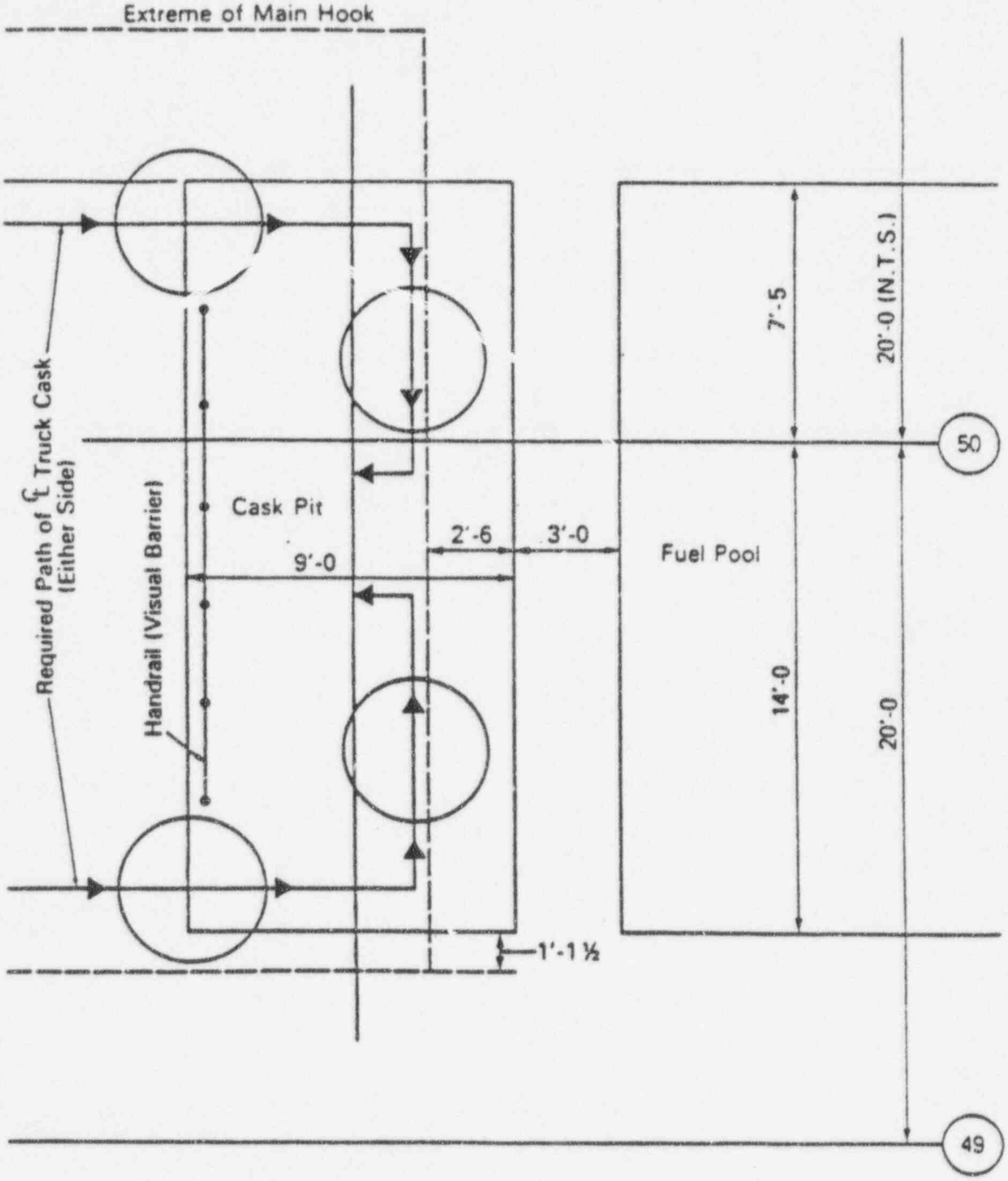

FIGURE 3.9-1
REQUIRED PATH FOR MOVEMENT OF TRUCK CASKS

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or 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 once per 12 hours, one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate:

a. greater than or equal to 1000 gpm; and

b. sufficient to maintain the RCS temperature at less than or equal to 140°F.

*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 once per 12 hours, one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate:

a. greater than or equal to 1000 gpm; and

b. sufficient to maintain the RCS temperature at less than or equal to 140°F.

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*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 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.10 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.11 FUEL HANDLING VENTILATION EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.11 The Fuel Handling Ventilation Exhaust System shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With the Fuel Handling Ventilation Exhaust System inoperable, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until the Fuel Handling Ventilation Exhaust System is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11.1 The Fuel Handling Ventilation Exhaust System shall be determined to be operating and discharging through the HEPA filters and charcoal adsorbers at least once per 12 hours whenever irradiated fuel is being moved in the storage pool and during crane operation with loads over the storage pool.

4.9.11.2 The above required Fuel Handling Ventilation Exhaust System shall be demonstrated OPERABLE:

a. 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 1% and uses the test procedure guidance of 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 35,000 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of 35,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

b. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

c. 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 inches Water Gauge while operating the system at a flow rate of 35,000 cfm ± 10%, and

2) Verifying that the exhaust flow rate is at least 8000 cfm greater than the supply flow rate.

d. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 35,000 cfm ± 10%; and

e. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 35,000 cfm ± 10%.

3/4.9.12 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.12 The boron concentration in the spent fuel pool shall be within the limit specified in the COLR.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Immediately suspend movement of fuel assemblies in the spent fuel pool and initiate action to restore the spent fuel pool boron concentration to within its limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 Verify at least once per 7 days that the spent fuel pool boron concentration is within its limit.

3/4.9.13 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 Storage of new or irradiated fuel is limited to the configurations described in this specification.

a. New or irradiated fuel may be stored in Region 1 of the Spent Fuel Pool in accordance with these limits:

   1) Unrestricted storage of fuel meeting the criteria of Table 3.9-1; or

   2) Restricted storage in accordance with Figure 3.9-1, of fuel which does not meet the criteria of Table 3.9-1.

b. New or irradiated fuel which has decayed at least 16 days may be stored in Region 2 of the Spent Fuel Pool in accordance with these limits:

   1) Unrestricted storage of fuel meeting the criteria of Table 3.9-3; or

   2) Restricted storage in accordance with Figure 3.9-2, of fuel which meets the criteria of Table 3.9-4; or

   3) Checkerboard storage in accordance with Figure 3.9-3 of fuel which does not meet the criteria of Table 3.9-4.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Immediately initiate action to move the noncomplying fuel assembly to the correct location.

b. The provisions of Specification 3.0.3 are not applicable.

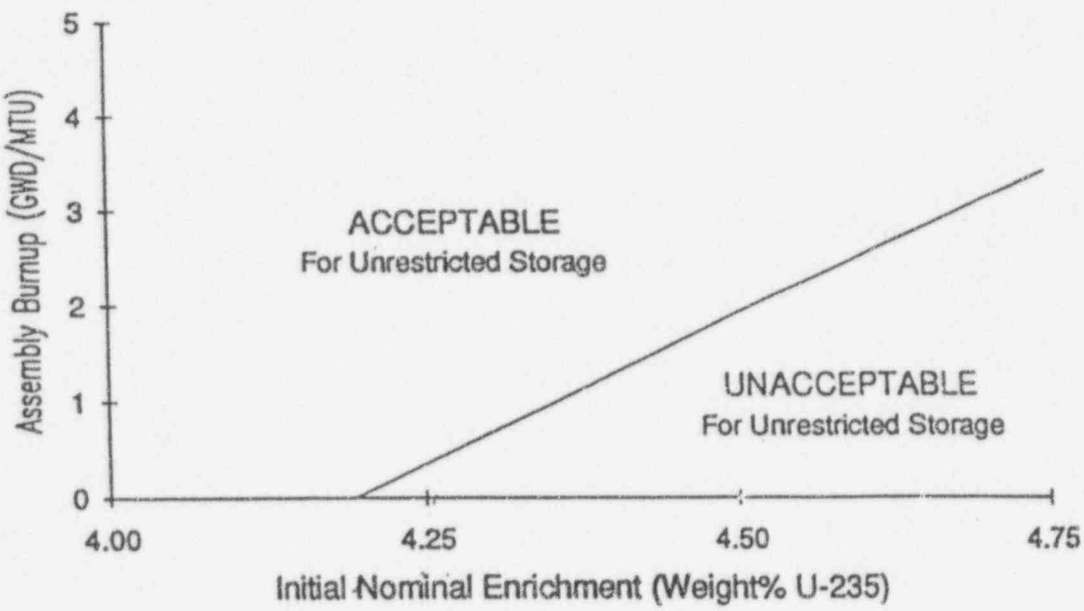
SURVEILLANCE REQUIREMENTS

4.9.13 Prior to storing a fuel assembly in the spent fuel storage pool, verify by administrative means the initial enrichment and burnup of the fuel assembly are in accordance with Specification 3.9.13.

Table 3.9-1

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Region 1 Storage

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
4.19(or less)	0
4.20	0.04
4.50	1.92
4.75	3.40



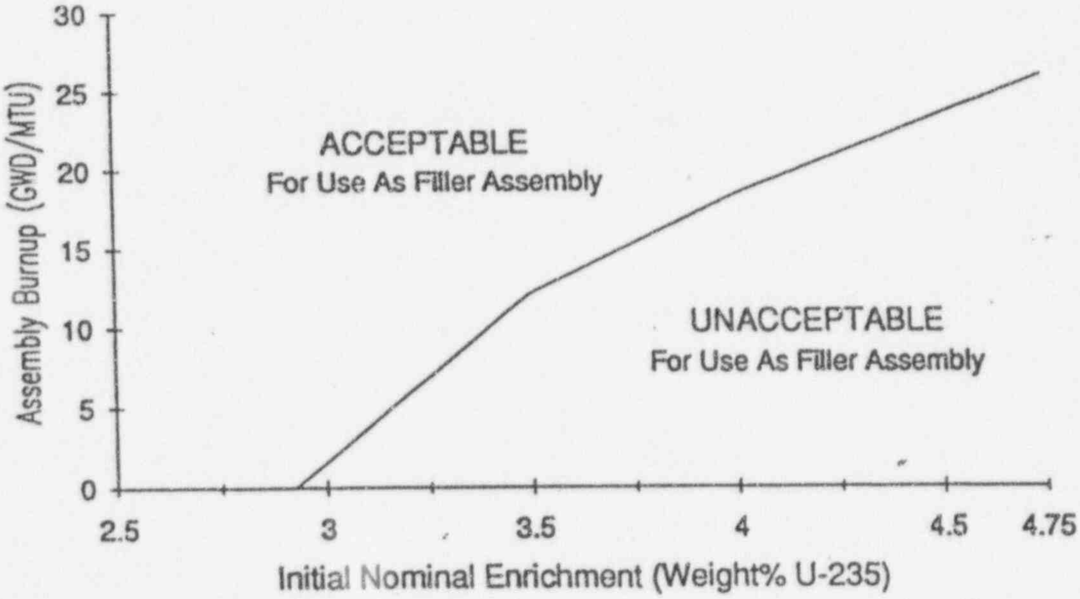
Fuel which differs from those designs used to determine the requirements of Table 3.9-1 may be qualified for Unrestricted Region 1 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Likewise, previously unanalyzed fuel up to 4.75 weight% U-235 may be qualified for Restricted Region 1 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-2

Minimum Qualifying Burnup Versus Initial Enrichment for Region 1 Filler Assemblies

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.92 (or less)	0
3.00	1.57
3.50	13.30
4.00	18.32
4.50	23.36
4.75	25.84

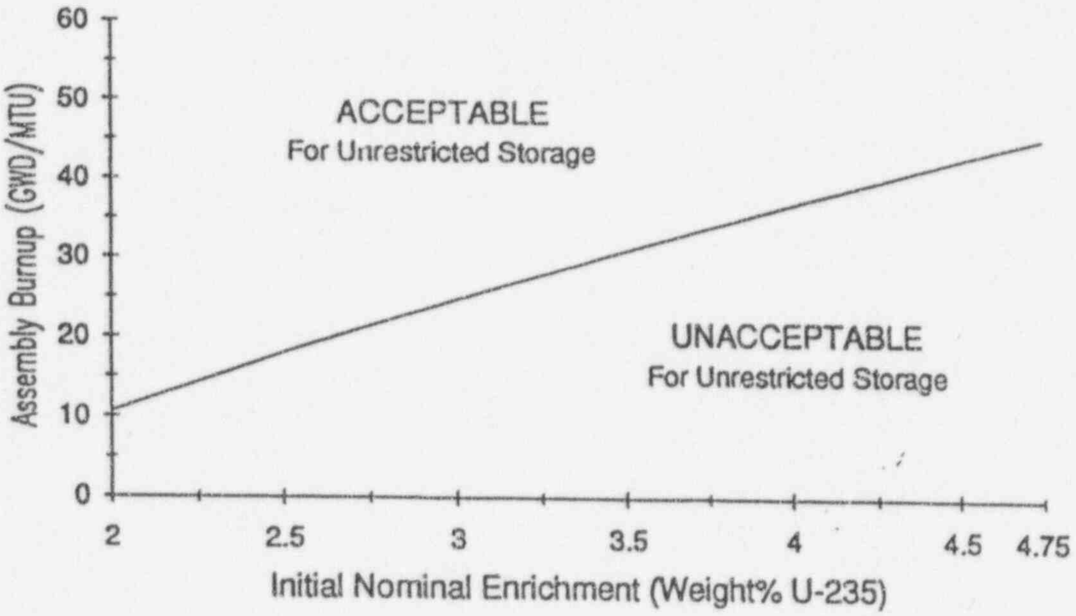


Fuel which differs from those designs used to determine the requirements of Table 3.9-2 may be qualified for use as a Region 1 Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-3

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Region 2 Storage

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00 (or less)	10.54
2.50	17.96
3.00	24.64
3.50	30.86
4.00	36.75
4.50	42.38
4.75	45.10

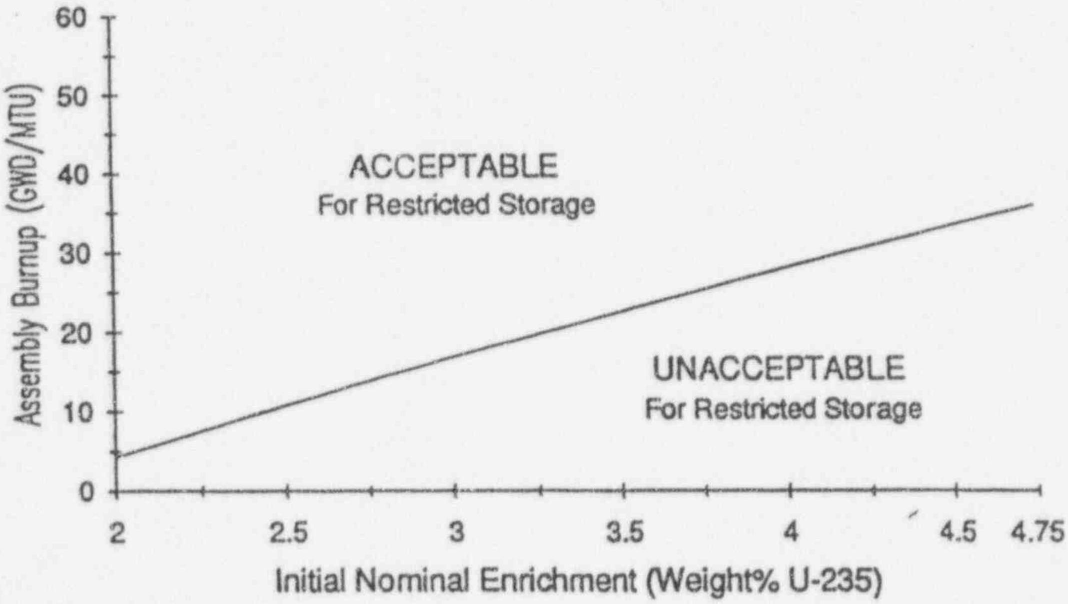


Fuel which differs from those designs used to determine the requirements of Table 3.9-3 may be qualified for Unrestricted Region 2 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-4

Minimum Qualifying Burnup Versus Initial Enrichment for Restricted Region 2 Storage with Fillers

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00 (or less)	4.22
2.50	10.75
3.00	16.80
3.50	22.41
4.00	27.92
4.5C	33.14
4.75	35.65

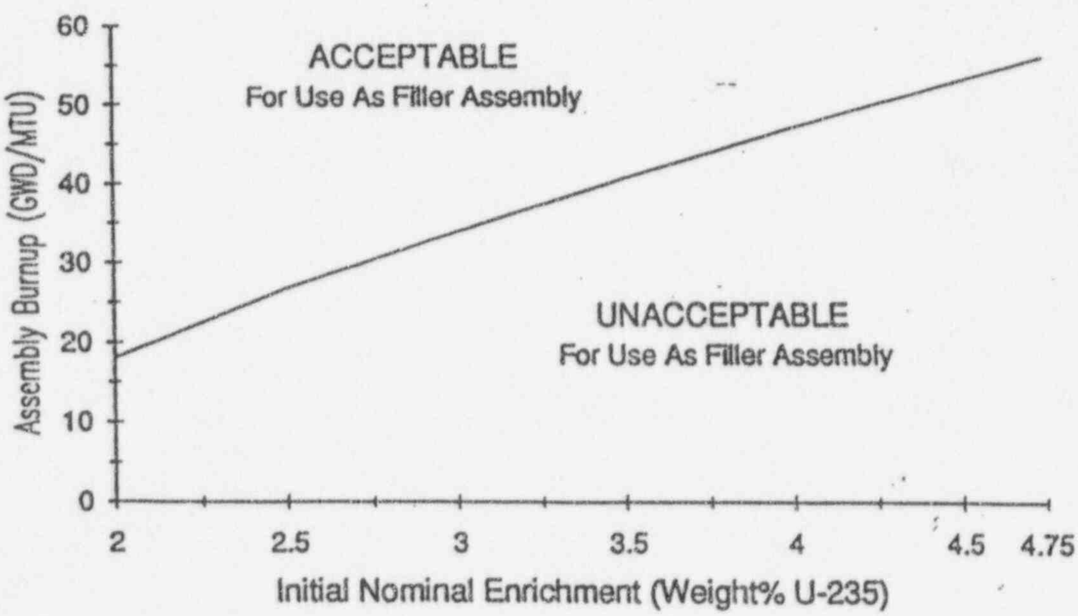


Fuel which differs from those designs used to determine the requirements of Table 3.9-4 may be qualified for Restricted Region 2 Storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-5

Minimum Qualifying Burnup Versus Initial Enrichment for Region 2 Filler Assemblies

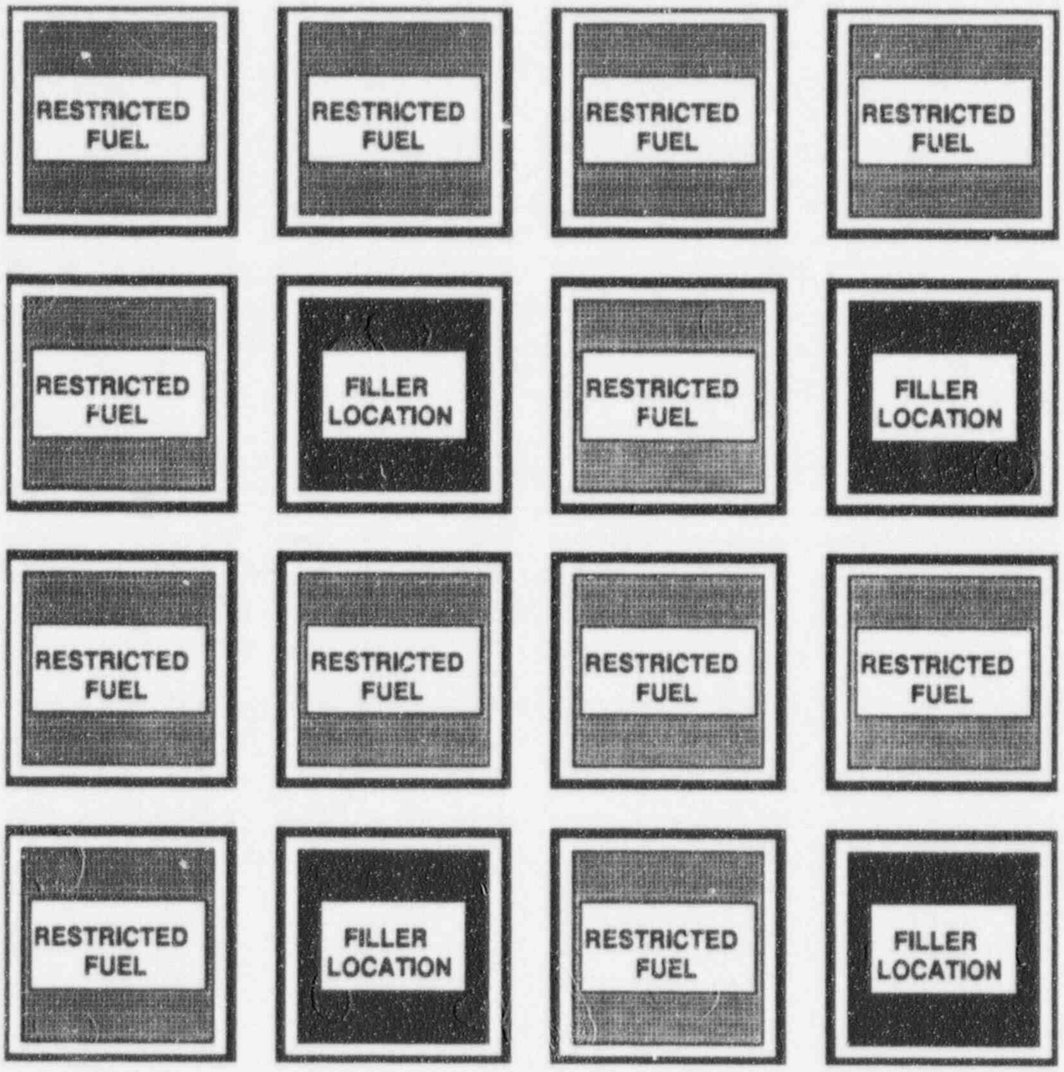
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00 (or less)	18.03
2.50	26.71
3.00	33.79
3.50	40.56
4.00	46.83
4.50	52.86
4.75	55.78



Fuel which differs from those designs used to determine the requirements of Table 3.9-5 may be qualified for use as a Region 2 Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Figure 3.9-1

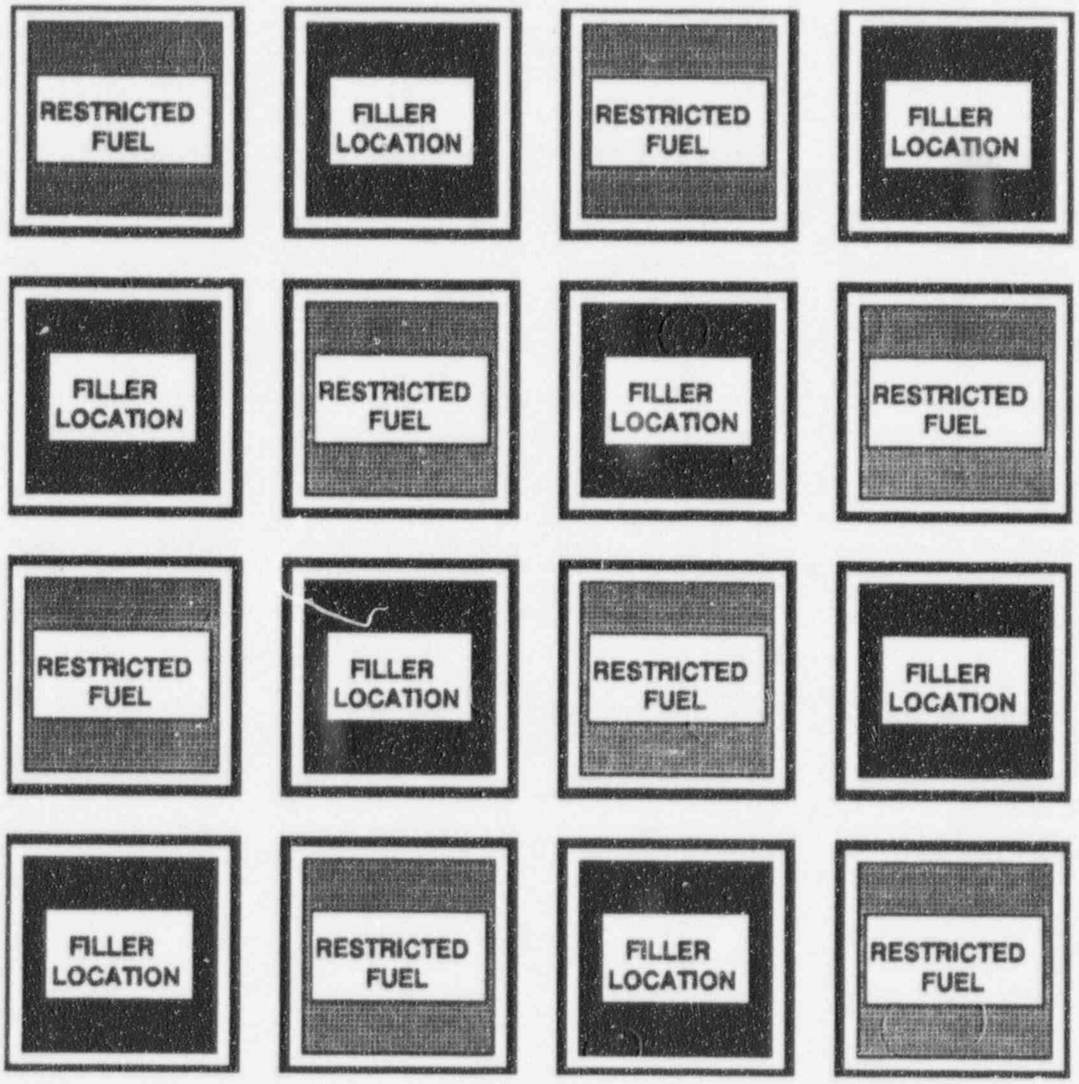
Required 3 out of 4 Loading Pattern for Restricted Region 1 Storage



Restricted Fuel:	Fuel which does <u>not</u> meet the minimum burnup requirements of Table 3.9-1. (Fuel which does meet the requirements of Table 3.9-1, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed)
Filler Location:	Either fuel which meets the minimum burnup requirements of Table 3.9-2, or an empty cell.
Boundary Condition:	Any row bounded by a Region 1 Unrestricted Storage Area shall contain a combination of restricted fuel assemblies and filler locations arranged such that no restricted fuel assemblies are adjacent to each other. Example: In the figure above, row 1 or column 1 can <u>not</u> be adjacent to a Region 1 Unrestricted Storage Area, but row 4 or column 4 can be.

Figure 3.9-2

Required 2 out of 4 Loading Pattern
for Restricted Region 2 Storage



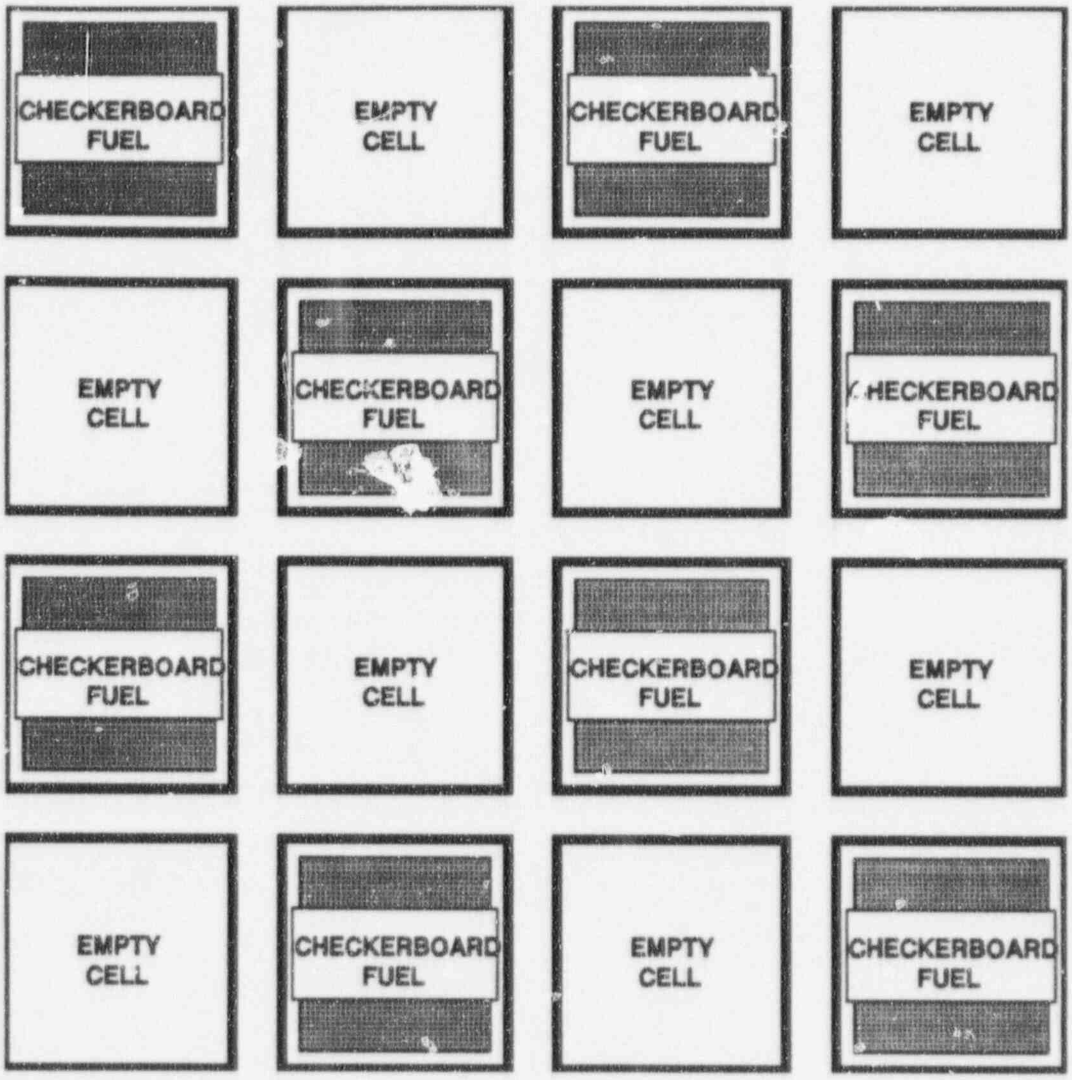
Restricted Fuel: Fuel which meets the minimum burnup requirements of Table 3.9-4, or non-fuel components, or an empty location.

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.9-5, or an empty cell.

Boundary Condition: No restrictions on boundary assemblies.

Figure 3.9-3

Required 2 out of 4 Loading Pattern for Checkerboard Region 2 Storage



Checkerboard Fuel: Fuel which does not meet the minimum burnup requirements of Table 3.9-4. (Fuel which does meet the requirements of Table 3.9-4, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed)

Boundary Condition: At least two opposite sides shall be bounded by either an empty row of cells, or a spent fuel pool wall.

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Nuclear channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

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, Power Range Channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided:

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Rod Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Rod Position Indication Systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

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*This requirement is not applicable during the initial calibration of the Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 DELETED

3/4.11.1.2 DELETED

3/4.11.1.3 DELETED

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each temporary unprotected outdoor tank shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

RADIOACTIVE EFFLUENTS

3/4 11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DELETED

3/4.11.2.2 DELETED

3/4.11.2.3 DELETED

3/4.11.2.4 DELETED

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, and immediately take ACTION a. above.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 49,000 Curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

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 Section 3 and 4 but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

BASES

Specifications 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of ACTION requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepared for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to

POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.4 establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

Specifications 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

BASES

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement 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 shall 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. These requirements apply except when relief has been provided in writing by the Commission.

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 the 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. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, tnat is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, 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 1.3% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections 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 End Of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value 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 EOL MTC value.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

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 pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) 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, (5) associated Heat Tracing Systems, and (6) 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 MARGIN from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions. The minimum borated water volumes and concentrations required to maintain shutdown margin for the Boric Acid Storage System and the Refueling Water Storage Tank are presented in the Core Operating Limits Report.

The Technical Specification LCO value for the Boric Acid Storage Tank and the Refueling Water Storage Tank minimum contained water volume during Modes 1-4 is based on the required volume to maintain shutdown margin, an allowance for unusable volume and additional margin as follows:

Boric Acid Storage Tank Requirements for Maintaining SDM - Modes 1-4

Required volume for maintaining SDM	presented in the COLR
Unusable volume (to maintain full suction pipe)	4,199 gallons
Additional margin	6,470 gallons

BORATION SYSTEMS (Continued)

Refueling Water Storage Tank Requirements for Maintaining SDM - Modes 1-4

Required volume for maintaining SDM	presented in the COLR
Unusable volume (below nozzle)	16,000 gallons
Additional margin	17,893 gallons

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 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Allowing two Centrifugal Charging pumps to operate simultaneously for ≤15 minutes increases the margin of safety with respect to the Reactor Coolant pump seal failure resulting in a LOCA in that the Reactor Coolant pump seal injection flow is not interrupted during pump swap. For the 15 minute period during which simultaneous Centrifugal Charging pump operation is allowed, the safety margins as related to the mass addition analysis are not appreciably reduced. Technical Specification 3.4.9.3 requires two PORVs to be operable during this period of operation, thus a mass addition transient can be relieved as required assuming the two PORVs function properly.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. The minimum borated water volumes and concentrations required to maintain shutdown margin for the Boric Acid Storage System and the Refueling Water Storage Tank are presented in the Core Operating Limits Report.

The Technical Specification LCO value for the Boric Acid Storage Tank and the Refueling Water Storage Tank minimum contained water volume during Modes 5 and 6 is based on the required volume to maintain shutdown margin, an allowance for unusable volume and additional margin as follows:

Boric Acid Storage Tank Requirements for Maintaining SDM - Modes 5 & 6

Required volume for maintaining SDM	presented in the COLR
Unusable volume (to maintain full suction pipe)	4,199 gallons
Additional margin	4,100 gallons

Refueling Water Storage Tank Requirements for Maintaining SDM - Modes 5 & 6

Required volume for maintaining SDM	presented in the COLR
Unusable volume (below nozzle)	16,000 gallons
Additional margin	6,500 gallons

BORATION SYSTEMS (Continued)

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 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The control rod insertion limit and shutdown rod insertion limits are specified in the CORE OPERATING LIMITS REPORT per specification 6.9.1.9.

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 LCO's are satisfied.

For Specification 3.1.3.1 ACTIONS c. and d., it is incumbent upon the plant personnel to verify the trippability of the inoperable control rod(s). 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.

MOVABLE CONTROL ASSEMBLIES (Continued)

During performance of the Control Rod Movement periodic test (Specification 4.1.3.1.2), there have been some "Control Malfunctions" that prohibited a control rod bank or group from moving when selected, as evidenced by the demand counters and DRPI*. In all cases, when the control malfunctions were corrected, the rods moved freely (no excessive friction or mechanical interference) and were trippable.

This surveillance test is an indirect method of verifying the control rods are not immovable or untrippable. It is highly unlikely that a complete control rod bank or bank group is immovable or untrippable. Past surveillance and operating history provide evidence of "trippability."

Based on the above information, during performance of the rod movement test, if a complete control rod bank or group fails to move when selected and can be attributed to a "Control Malfunction," the control rods can be considered "Operable" and plant operation may continue while ACTIONS c. and d. are taken.

If one or more control rods fail to move during testing (not a complete bank or group and cannot be contributed to a "Control Malfunction"), the affected control rod(s) shall be declared "Inoperable" and ACTION a. taken.

(Reference: W letter dated December 21, 1984, NS-NRC-84-2990, E. P. Rahe to Dr. C. O. Thomas)

*Digital Rod Position Indicators

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 calculated DNBR in the core at or above the design limit 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 are not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(X,Y,Z)$ Heat Flux Hot Channel Factor, is defined as the local heat flux on the surface of a fuel rod at core location X,Y,Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}(X,Y)$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along a rod at core location X,Y to the average rod power.

K(Z) is defined as the normalized $F_Q(X,Y,Z)$ limit for a given core height.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) ensure that $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ limits specified in the CORE OPERATING LIMITS REPORT (COLR) are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD envelope specified in the COLR has been adjusted for measurement uncertainty.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed ΔI power operating space during normal power operation. These alarms are active when power is greater than 50% of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, 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 ECCS acceptance criteria are not exceeded. The peaking limits are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.9.

The heat flux hot channel factor and nuclear enthalpy rise hot channel factor are each measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

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.

$F_{\Delta H}(X,Y)$ will be maintained within its limits provided Conditions a. through d. above are maintained.

The limits on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, are specified in the COLR as Maximum Allowable Radial Peaking (MARP) limits, obtained by dividing the Maximum Allowable Total Peaking (MAP) limit by the axial peak [AXIAL(X,Y)] for location (X,Y). By definition, the Maximum Allowable Radial Peaking limits will result in a DNBR for the limiting transient that is equivalent to the DNBR calculated with a design $F_{\Delta H}(X,Y)$ value of 1.50 and a limiting reference axial power shape. For transition cores, MARP limits may be applied to both MARK-BW and optimized fuel types provided allowances for differences in DNBR are accounted for in the generation of MARP limits. The MARP limits specified in the COLR include allowances for mixed core DNBR effects.

The relaxation of $F_{\Delta H}(X,Y)$ as a function of THERMAL POWER allows for a change in the radial power shape for all permissible control bank insertion limits. This relaxation is implemented by the application of the following factors:

$k = [1 + (1/RRH)\ (1 - P)]$

where k = power factor multiplier applied to the MAP limits
p = THERMAL POWER/RATED THERMAL POWER
RRH is given in the COLR

HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The hot channel factor $F^M_Q(X,Y,Z)$, and the nuclear enthalpy rise hot channel factor, $F^M_{\Delta H}(X,Y)$, are measured periodically to verify that the core is operating as designed. $F^M_Q(X,Y,Z)$ and $F^M_{\Delta H}(X,Y)$ are compared to allowable limits to provide reasonable assurance that limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety Systems Settings), 3.1.3 (Movable Control Assemblies), 3.2.1 (Axial Flux Difference), and 3.2.4 (Quadrant Power Tilt Ratio). A peaking margin calculation is performed to provide the basis for decreasing the width of the AFD and $f(\Delta I)$ limits and for reducing THERMAL POWER.

When an $F^M_Q(X,Y,Z)$ measurement is obtained from a full-core map in accordance with surveillance requirements of Specification 4.2.2, no uncertainties are applied to the measured peak since a measurement uncertainty of 5.0% and a manufacturing tolerance of 3.0% are included in the peaking limit. When $F^M_Q(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2, the measured peak is increased by the radial-local peaking factor and appropriate allowances for measurement uncertainty and for manufacturing tolerances.

When an $F^M_{\Delta H}(X,Y)$ measurement is obtained from a full-core map, regardless of the reason, no uncertainties are applied to the measured peak since the required uncertainties are included in the peaking limit.

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 the x-y plane power tilts. The peaking increase that corresponds to a QUADRANT POWER TILT RATIO of 1.02 is included in the generation of the AFD limits.

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 rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q(X,Y,Z)$ is reinstated by reducing the power by 3% from RATED THERMAL POWER for each percent of tilt in excess of 2.0%.

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.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design limit DNBR throughout each analyzed transient. As noted on Figure 3.2-1, RCS flow rate and THERMAL POWER may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the power level is decreased) to ensure that the calculated DNBR will not be below the design DNBR value. The relationship defined on Figure 3.2-1 remains valid as long as the limits placed on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, in Specification 3.2.3 are maintained. The indicated T_{avg} values and the indicated pressurizer pressure values correspond to analytical limits of 592.6°F and 2220 psia respectively, with allowance for indication instrumentation measurement uncertainty. When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-1 since an RCS total flow rate measurement uncertainty, greater than or equal to the value stated on Figure 3.2-1 has been allowed for in determination of the design DNBR value.

3/4.2.5 DNB PARAMETERS (Continued)

The measurement error for RCS total flow rate is based upon the performance of past precision heat balances. Sets of elbow tap coefficients, as determined during these heat balances, were averaged for each elbow tap to provide a single set of elbow tap coefficients for use in calculating RCS flow. This set of coefficients establishes the calibration of the RCS flow rate indicators and becomes the set of elbow tap coefficients used for RCS flow measurement. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from these heat balances in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-1. 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 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. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip and Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features Instrumentation and (3) sufficient system functions 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 accident analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. The NRC Safety Evaluation Reports for the WCAP-10271 series were provided in letters dated February 21, 1985 from C. O Thomas (NRC) to J. J. Sheppard (WOG), February 22, 1989 from C. E. Rossi (NRC) to R. A. Newton (WOG), and April 30, 1990 from C. E. Rossi (NRC) to G. T. Goering (WOG).

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Feature actuation associated with each channel is completed within the time limit assumed in the accident 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 times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, and (10) nuclear service water pumps start and automatic valves position.

Technical Specifications for the Reactor Trip Breakers and the Reactor Trip Bypass Breakers are based upon NRC Generic Letter 85-09 "Technical Specifications for Generic Letter 83-28, Item 4.3," dated May 23, 1985.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 Defeats the manual block of Safety Injection actuation on low pressurizer pressure and low steamline pressure and defeats steamline isolation on negative steamline pressure rate. Defeats the manual block of the motor-driven auxiliary feedwater pumps on trip of main feedwater pumps and low-low steam generator water level.

P-12 On 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 level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 Deleted

3/4.3.3.4 Deleted

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY 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 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations "

3/4.3.3.7 DELETED

3/4.3.3.8 NOT USED

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The gas instrumentation is provided for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM.

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat; however, single failure considerations require that three loops be OPERABLE. Also, the uncontrolled bank withdrawal from zero power or subcritical assumes three reactor coolant loops in operation.

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 a reactor coolant pump with one or more RCS cold legs less than or equal to 300°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.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the 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 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no Reactor trip until the first Reactor Trip System 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. Specifications 3.4.2.1 and 3.4.2.2 allow a + 3% and - 2% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during surveillance testing to allow for drift.

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. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions: 1) Manual control of PORVs to control RCS pressure. This is a function that is used for the steam generator tube rupture accident coincident with a loss of all offsite power and for plant shutdown. 2) Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified

3/4.4.4 RELIEF VALVES (Continued)

reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of RCS pressure and isolate a PORV with excessive leakage. 4) Automatic control of PORVs to control RCS pressure. This is a function that reduces challenges to the code safety valves for overpressurization events. 5) Manual control of a block valve to isolate a stuck-open PORV.

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 B&W process (or method) equivalent to the inspection method described in Topical Report BAW-2045(P)-A will be used. Inservice inspection of steam generator sleeves is also required to ensure RCS integrity. Because the sleeves introduce changes in the wall thickness and diameter, they reduce the sensitivity of eddy current testing, therefore, special inspection methods must be used. A method is described in Topical Report BAW-2045(P)-A with supporting validation data that demonstrates the inspectability of the sleeve and underlying tube. As required by NRC for licensees authorized to use this repair process, McGuire commits to validate the adequacy of any system that is used for periodic inservice inspections of the sleeves, and will evaluate and, as deemed appropriate by Duke Power Company, implement testing methods as better methods are developed and validated for commercial use.

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

STEAM GENERATORS (Continued)

will be found during scheduled inservice steam generator tube examinations. Repair will be required for all tubes with imperfections exceeding the repair limit of 40% of the tube nominal wall thickness. Installed sleeves with imperfections exceeding 40% of the sleeve nominal wall thickness will be plugged. Defective steam generator tubes can be repaired by the installation of sleeves which span the area of degradation, and serve as a replacement pressure boundary for the degraded portion of the tube, allowing the tube to remain in service. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness. For tubes with degradation below the F* distance, and not degraded within the F* distance, repair is not required. If a tube is sleeved due to degradation in the F* distance, then any defects in the tube below the sleeve will remain in service without repair.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73 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

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 Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

OPERATIONAL LEAKAGE (Continued)

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 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the accident analyses.

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.

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.

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 Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm. The values for the limits on specific activity represent limits based upon a parametric

SPECIFIC ACTIVITY (Continued)

evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the McGuire 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 1.0 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.

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 micro-Curie/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 radio-iodine contribution would probably be about 20%. The exclusion of radio-nuclides 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 radionu-clides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective ACTION. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of the effective full power years (EFPY) of service life identified on the applicable technical specification figure. The 16 EFPY service life period continues to ensure that the limiting RT_{NDT} at the 1/4T location in the core region is a bounding value. 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 phosphate content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT}. 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 the identified service life. Adjustments for possible errors in the pressure and temperature sensing instruments are included when stated on the applicable figure.

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 50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the pressure vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the pressure vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS

COMPONENT	CU (%)	P (%)	NDTT (°F)	MINIMUM 50 FT-LB/35 MIL TEMP (°F) PARALLEL TO MAJOR WORKING DIRECTION	MINIMUM 50 FT-LB/35 MIL TEMP (°F) NORMAL TO MAJOR WORKING DIRECTION(b)	RT_{NDT} (°F)	AVERAGE UPPER SHELF (FT-LB) PARALLEL TO MAJOR WORKING DIRECTION	AVERAGE UPPER SHELF (FT-LB) NORMAL TO MAJOR WORKING DIRECTION(b)
Cl.Hd. Dome	-	-	-31	52	72	12	132	86
Cl. Hd.Ring	-	-	16	3	23	16	156	101.5
Hd. Flange	-	-	-13	41	61	1	155	100.5
Vessel Flange	-	-	- 4	21	41	- 4	174	113
Inlet Nozzle	-	-	-13	- 4	16	-13	141	92
Inlet Nozzle	-	-	-31	3	23	-31	114.5	74.5
Inlet Nozzle	-	-	-22	- 8	12	-22	129	84
Inlet Nozzle	-	-	-40	- 6	14	-40	132	86
Outlet Nozzle	-	-	-13	33	53	- 7	124	81
Outlet Nozzle	-	-	-40	16	36	-24	103(c)	67(c)
Outlet Nozzle	-	-	-49	24	44	-16	116(c)	75.5(c)
Outlet Nozzle	-	-	-40	10	30	-30	121(c)	78.5(c)
Upper Shell	-	-	- 4	41	61	1	157.5	102.5
Inter. Shell	0.16	0.012	- 4	7	37(a)	- 4(a)	147	96(c)
Lower Shell	0.15	0.004	-30	-22	7(a)	-30(a)	152	141(a)
Bot. Hd. Ring	-	-	- 4	55	75	15	109(c)	71(c)
Bot. Hd. Peel	-	-	-49	38	58	- 2	136	88.5
Bot. Hd. Peel	-	-	-40	-24	- 4	-40	131	85
Bot. Hd. Peel	-	-	-13	-18	2	-13	142	92
Bot. Hd. Peel	-	-	-13	- 8	12	-13	132	86
Bot. Hd. Dome	-	-	-40	-12	8	-40	127	82.5
Weld(Inter/Lower)	0.05	0.010	-76	NA	- 8(a)	-68(a)	N.A.	128(a)
HAZ	-	-	-76	NA	-58(a)	-76(a)	N.A.	125.5(a)

(a) Based on Actual Data
(b) Estimated Per Branch Technical Position MTEB 5-2
(c) 100% Shead not reached; Upper Shelf Toughness will be greater than that listed.

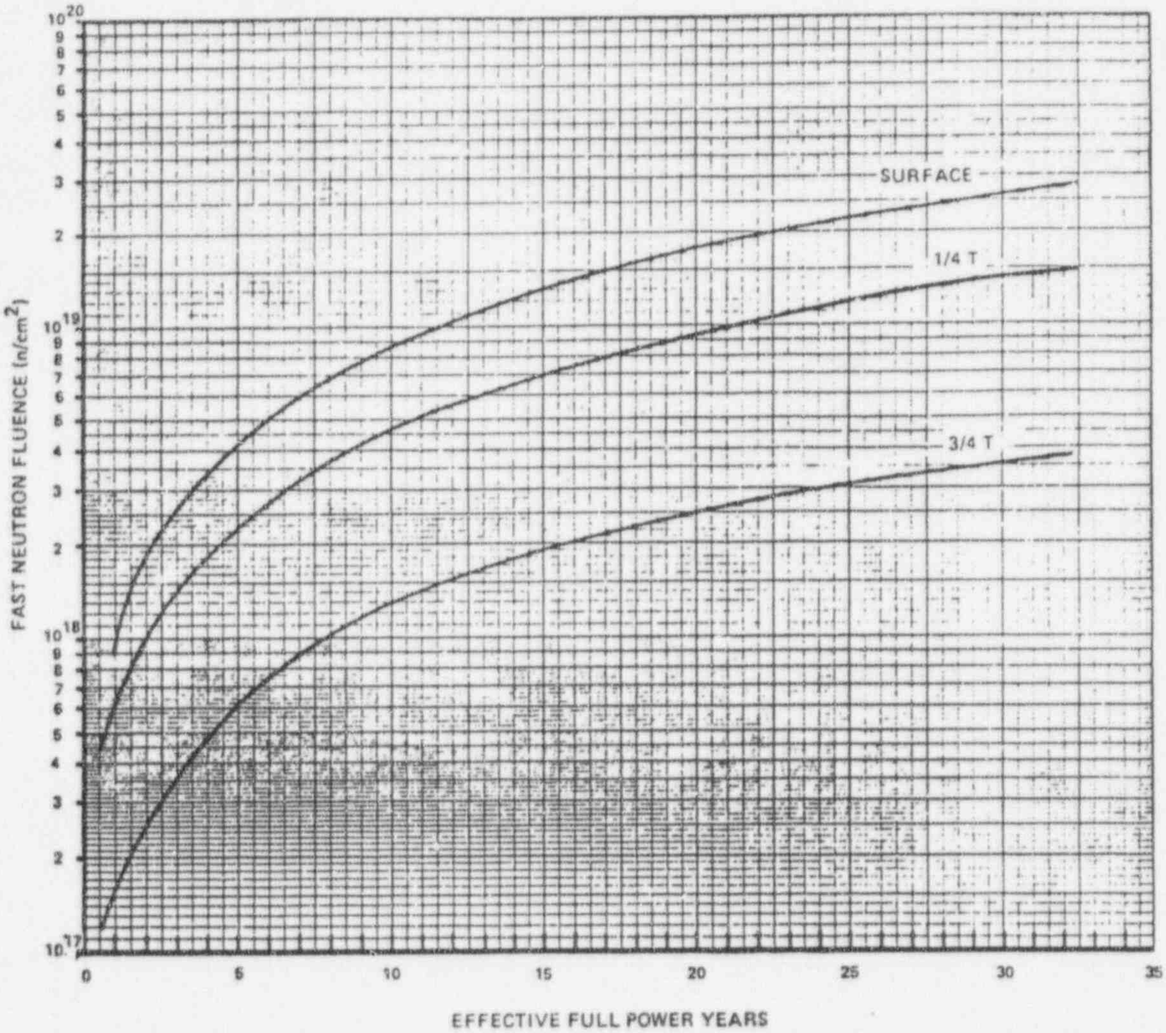

FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E>1MEV)
AS A FUNCTION OF EFFECTIVE FULL POWER YEARS

PRESSURE/TEMPERATURE LIMITS (Continued)

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical 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 non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

Where: K_{IM} is the stress intensity factor caused by membrane (pressure) stress,

K_{IT} is the stress intensity factor caused by the thermal gradient

K_{IR} is 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

PRESSURE/TEMPERATURE LIMITS (Continued)

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 delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T 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

PRESSURE/TEMPERATURE LIMITS (Continued)

end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves in technical specifications for the heatup rate data and the cooldown rate data may be adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves. Where technical specification curves have not been adjusted, such adjustments are made by plant procedures.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.75 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 300°F. Either of

PRESSURE/TEMPERATURE LIMITS (Continued)

the PORVs or the RCS vent opening 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 HPSI pump and its injection into a water-solid RCS. The Pressurizer PORV setpoints for low temperature overpressure protection are based on limiting the peak pressure during the limiting transient to 1.10 times the ASME Section XI, Appendix G limits, in accordance with ASME code case N-514.

Credit is taken for the RHR suction relief valve (ND-3) during conditions where relieving capacity at rated accumulation is sufficient to prevent exceeding the above allowable pressure limits.

Cooldown limits/minimum RCS temperature restrictions ensure the allowable pressure limits will not be exceeded.

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 Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 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, 1971 Edition and Addenda through Winter 1972.

3/4.4.11 REACTOR VESSEL HEAD VENT SYSTEM

Reactor Vessel Head Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function. (Operability of the pressurizer steam space vent path is provided by Specifications 3/4.4.4 and 3/4.4.9.3.)

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 surveillance to verify Reactor Vessel Head Vent flowpath is qualitative as no specific size or flow rate is required to exhaust noncondensible gases. The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) Cold Leg 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 allowed outage time for the accumulators are variable based upon boron concentration to ensure that the reactor is shutdown following a LOCA and that any problems are corrected in a timely manner. The minimum boron concentration required to ensure post-LOCA subcriticality as presented in the Core Operating Limits Report, is based on nominal accumulator volume conditions and allows additional outage time since subcriticality is assured when the boron concentration is above this value. A slightly high boron concentration, the minimum accumulator boron concentration limit for LCO 3.5.1c presented in the Core Operating Limits Report, is based on worst case liquid mass, boron concentration and measurement errors. A concentration less than this LCO value in any single accumulator or as a volume weighted average may be indicative of a problem, such as valve leakage. Since reactor shutdown is assured if the boron concentration is above the minimum concentration to ensure post-LOCA subcriticality and the accumulator volume is greater than or equal to the nominal volume, additional time is allowed to restore boron concentration in the accumulators.

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.

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 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Allowing two Centrifugal Charging pumps to operate simultaneously for ≤15 minutes increases the margin of safety with respect to the Reactor Coolant pump seal failure resulting in a LOCA in that the Reactor Coolant pump seal injection flow is not interrupted during pump swap. For the 15 minute period during which simultaneous Centrifugal Charging pump operation is allowed, the safety margins as related to the mass addition analysis are not appreciably reduced. Technical Specification 3.4.9.3 requires two PORVs to be operable during this period of operation, thus a mass addition transient can be relieved as required assuming the two PORVs function properly.

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 [Deleted]

3/4.5.5 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 7.5 and 9.5 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.

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 accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 1.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 15 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 14.5 psig. The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 14.8 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that: (1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions, and (2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 100°F for the lower compartment, 75°F for the upper compartment and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to 14.8 psig which is less than the containment design pressure of 15 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 15 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 REACTOR BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment reactor building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide: (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

3/4.6.1.8 ANNULUS VENTILATION SYSTEM

The OPERABILITY of the Annulus Ventilation System ensures that during LOCA conditions, containment vessel leakage into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The specified laboratory test method, ASTM-3803-89, implies that heaters may be unavailable for controlling the relative humidity of the influent air entering the charcoal absorber section to ≤ 70 percent. This is acceptable since the accident analysis with appropriate absorber efficiencies for radioiodine in elemental and organic forms based on the above test shows that the control room radiation doses to be within the 10 CFR Part 50, Appendix A, GDC 19 limits during design basis LOCA conditions. However, specifications are included to ensure heater operability and corrections ACTIONS are identified to address the contingency of inoperable heaters; these are in place to increase the safety margin of the filters. Heater operation is not necessary to meet the assumptions used in the accident analyses and limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during LOCA conditions.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves for the lower compartment (24-inch) and instrument room (12-inch and 24-inch) are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the purge supply and exhaust isolation valves in the upper compartment (24-inch) since, unlike the valves in the lower compartment and instrument room, the upper compartment valves will close during a LOCA. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation with these valves open will be limited to 250 hours during a calendar year.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM (Continued)

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

CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

Containment isolation valves are listed in FSAR Table 6.2.4-1. Those valves with a required isolation time have a value given in the "MAX ISOLATION TIME (SEC)" column. Penetration test type (type B, type C, or None) is listed in the "TEST TYPE" column of the table for each containment penetration. Changes to the FSAR are made in accordance with 10 CFR 50.59.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

3/4.6.4 COMBUSTIBLE GAS CONTROL (Continued)

The OPERABILITY of at least 64 of 66 igniters ensures that the Distributed Ignition System will maintain an effective coverage throughout the containment. This system of igniters will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 14.8 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will: (1) be distributed evenly through the containment bays, (2) contain sufficient boron to preclude dilution of the containment sump following the LOCA, and (3) contain sufficient heat removal capability to condense the Reactor Coolant System volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1081 pounds of ice per basket contains a 10% conservative allowance for ice loss through sublimation which is a factor of 10 higher than assumed for the ice condenser design. The minimum weight figure of 2,099,790 pounds of ice also contains an additional 1.1% conservative allowance to account for systematic error in weighing instruments. In the event that observed sublimation rates are equal to or lower than design predictions after 3 years of operation, the minimum ice baskets weight may be adjusted downward. In addition, the number of ice baskets required to be weighed each 9 months may be reduced after 3 years of operation if such a reduction is supported by observed sublimation data.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the Ice Bed Temperature Monitoring System ensures that the capability is available for monitoring the ice temperature. In the event the system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.3 ICE CONDENSER DOORS

The OPERABILITY of the ice condenser doors and the requirement that they be maintained closed ensures that the Reactor Coolant System fluid released during a LOCA will be diverted through the ice condenser bays for heat removal and that excessive sublimation of the ice bed will not occur because of warm air intrusion.

If an ice condenser door is not capable of opening automatically, then system function is seriously degraded and immediate action must be taken to restore the opening capability of the door. Not capable of opening automatically is defined as those conditions in which a door is physically blocked from opening by installation of a blocking device or by obstruction from temporary or permanent installed equipment or is otherwise inhibited from opening such as may result from ice, frost, debris or increased door opening torque.

3/4.6.5.4 INLET DOOR POSITION MONITORING SYSTEM

The OPERABILITY of the Inlet Door Position Monitoring System ensures that the capability is available for monitoring the individual inlet door position. In the event the system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.5 DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

The requirements for the divider barrier personnel access doors and equipment hatches being closed and OPERABLE ensure that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of the steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.6.5.6 CONTAINMENT AIR RETURN AND HYDROGEN SKIMMER SYSTEM

The OPERABILITY of the Containment Air Return and Hydrogen Skimmer System ensures that following a LOCA: (1) the containment atmosphere is circulated for cooling by the Spray System, and (2) the accumulation of hydrogen in localized portions of the containment structure is minimized.

3/4.6.5.7 and 3/4.6.5.8 FLOOR AND REFUELING CANAL DRAINS

The OPERABILITY of the ice condenser floor and refueling canal drains ensures that following a LOCA, the water from the melted ice and Containment Spray System has access for drainage back to the containment lower compartment and subsequently to the sump. This condition ensures the availability of the water for long-term cooling of the reactor during the post-accident phase.

3/4.6.5.9 DIVIDER BARRIER SEAL

The requirement for the divider barrier seal to be OPERABLE ensures that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary Coolant System pressure will be limited to within 110% 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. Table 3.7-3 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during surveillance testing to allow for drift. The total relieving capacity for all valves on all of the steam lines is 15.9×10^6 lbs/hr which is 105% of the total secondary steam flow of 15.14×10^6 lbs/hr 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 Tables 3.7-1 and 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:

$$SP = \frac{(X) - (Y)(U)}{X} \times (*)$$

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,

* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for three loop operation. This value left blank pending NRC approval of 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 conditions in the event of a total loss-of-offsite power.

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1210 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 900 gpm at a pressure of 1210 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

Verification of the steam turbine-driven pump discharge pressure should be deferred until suitable test conditions are established (i.e., greater than or equal to 900 psig in the secondary side of the steam generator). This deferral is required because until 900 psig is reached, there is insufficient steam pressure to perform the test.

3/4.7.1.3 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.4 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the accident 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 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

The OPERABILITY of the Nuclear Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits. Periodic flow balance tests, delta-P tests, and heat balance tests are performed as required to assure adequate flow to all essential heat exchangers for which flow instrumentation is provided.

Portions of the Nuclear Service Water System are common to both units. These shared portions of the system are indicated on Figure 3/4 7-1 and common valves are listed in Table B 3/4 7-1 and include common suction piping and cross-connect piping as indicated on the figure.

3/4.7.4 NUCLEAR SERVICE WATER SYSTEM (Continued)

With the exception of ORN-1 all shared valves receive emergency power from two essential motor control centers (1EMXH, 2EMXH). ORN-1 is normally open with the power removed. Motor Control Center (MCC) 1EMXH can be powered by either Unit 1 or Unit 2 A Train Emergency D/G's via their associated switchgear. Motor Control Center 2EMXH can be powered by either Unit 1 or 2 B Train Emergency D/G's via their associated switchgear.

The four loops (two per unit) ensure redundancy and the availability of cooling to both units, even if a single failure were to render two loops inoperable. (Such a failure would involve train A of both units or train B of both units, not both trains of the same unit). The Action statements are separated to clarify that portions of the systems are shared though the majority of each of the four loops is independent.

In the event of a safety injection or blackout signal on either unit, train A of both units will align to Lake Norman and train B of both units will align to the SNSWP. Additionally, all train A to train B cross-connects will close on both units as will non-safety to safety related cross-connects. These actuations assure independence of the loops and the required redundancy under design basis conditions.

3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND

The limitations on the standby nuclear service water pond level and temperature ensure that sufficient cooling capacity is available either to: (1) provide normal cooldown of the facility, or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974. The Surveillance Requirements specified for the dam inspection will conform to the recommendations of Regulatory Guide 1.127, Revision 1, March 1978

TABLE B 3/4 7-1

UNITS 1 AND 2
NUCLEAR SERVICE WATER SYSTEM SHARED VALVES

ORN-1	LOW LEVEL INTAKE SUP TO RN
ORN-2B	TRAIN A RC SUPPLY
ORN-3A,C	TRAIN A RC SUPPLY
ORN-4A,C	TRAIN B RC SUPPLY
ORN-5B	TRAIN B RC SUPPLY
ORN-7A,C	TRAIN A SNSWP SUPPLY
ORN-9B	TRAIN B SNSWP SUPPLY
ORN-10A,C	TRAIN B LLI SUPPLY
ORN-11B	TRAIN B LLI SUPPLY
ORN-12A,C	TRAIN A LLI SUPPLY
ORN-13A	TRAIN A LLI SUPPLY
ORN-14A	TRAIN A SUCT X-CONNECT
ORN-15B	TRAIN B SUCT X-CONNECT
ORN-147A,C	TRAIN A DISCH TO RC
ORN-148A,C	TRAIN A DISCH TO RC
ORN-149A	TRAIN A DISCH TO RC
ORN-150A,C	TRAIN A DISCH X-CONNECT
ORN-151B	TRAIN B DISCH X-CONNECT
ORN-152B	TRAIN B DISCH TO SNSWP
ORN-283A,C	TRAIN B DISCH TO RC
ORN-284B	TRAIN B DISCH TO RC
ORN-301A,C	RV SUPPLY FROM LLI
ORN-302B	RV SUPPLY FROM LLI

3/4.7.6 CONTROL AREA VENTILATION SYSTEM

The OPERABILITY of the Control Area Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The specified laboratory test method, namely, ASTM D3803-89, implies that heaters may be unavailable for controlling the relative humidity of the influent air entering the charcoal absorber section to ≤ 70 percent. This is acceptable since accident analysis with appropriate absorber efficiencies for radioiodine in elemental and organic forms based on the above test shows the site boundary radiation doses to be within the 10 CFR Part 100 limits during design basis LOCA conditions. However, specifications are included to ensure heater operability and corrective ACTIONS are identified to address the contingency of inoperable heaters; these are in place to increase the safety margin of the 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 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR 50. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.7 AUXILIARY BUILDING FILTERED VENTILATION EXHAUST SYSTEM

The OPERABILITY of the Auxiliary Building Filtered Ventilation Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the auxiliary building following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations were assumed in the accident analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. The methyl iodide penetration test criterion for the carbon samples has been established at 10% (i.e., 90% removal) which is greater than the iodine removal in the accident analysis.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program 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.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2 kip, 10 kip, and 100 kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured Company "B" for the purposes of this specification would be of a different type, as would hydraulic snubbers from either manufacturer.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide assurance of snubber functional reliability one of the three sampling and acceptance criteria methods are used:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or continue testing* 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 snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date.

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*If testing continues to between 100-200 snubbers (or 1-2 weeks) and still the accept region has not been reached, then the actual % of population quality (C/N) should be used to prepare for extended or 100% testing.

SNUBBERS (Continued)

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, 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. The requirements for the maintenance of records and the snubber service life review not intended to affect plant operation.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 Deleted

3/4.7.11 Deleted

3/4.7.12 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 3.9°F.

3/4.7.13 GROUNDWATER LEVEL

This Technical Specification is provided to ensure that groundwater levels will be monitored and prevented from rising to the potential failure limit for the McGuire Units 1 and 2 Auxiliary Buildings. This potential failure limit is based on engineering calculations that have determined that the Auxiliary Buildings are susceptible to overturning due to buoyancy at elevation 737 feet Mean Sea Level (MSL). Under the requirements of this Technical Specification, if groundwater level exceeds elevation 731 feet MSL, (3 out of 5 Tech Spec

GROUNDWATER LEVEL (Continued)

groundwater monitor alarms), and cannot be reduced in one (1) hour, McGuire must begin reducing Units 1 and 2 to Mode 5, Cold Shutdown.

Analysis performed by Design Engineering determined that the Reactor and Diesel Generator Buildings are designed to withstand hydrostatic loadings due to groundwater levels up to elevation 760 feet MSL; therefore, no Technical Specification requirements are specified for these structures.

Elevation 731 feet MSL is the Technical Specification action level of the five Technical Specification groundwater monitors listed in Table 3.7-7. The East Wall exterior monitor alarm at elevation 731 feet MSL is the Alert alarm. The other four (4) monitors are Hi-Hi alarms at elevation 731 feet (MSL).

The East Wall exterior monitor was originally on the exterior of the Unit 2 Auxiliary Building and subsequently was enclosed by the construction of the Equipment Staging Building.

As required by Operations procedures, any alarms on non-Technical Specification groundwater monitors will also be investigated. Additionally, if three (3) out of the five (5) Technical Specification groundwater monitors alarm at levels below the Technical Specification action levels, Operations will contact Duke Engineering (Civil) for investigation and resolution of the increased groundwater level.

If one or more of the 5 Technical Specification groundwater monitors is determined to be inoperable, the monitor(s) will be considered to be indicating above the 731'-0" level until repaired and returned to an operable status.

3/4.8.1, 3/4.8.2 AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

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 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 ACTION requirements for diesel generator testing in the event of the inoperability of other electric power sources also reflect the potential for degradation of the diesel generator due to excessive testing. This concern has developed, concurrently with increased industry experience with diesel generators, and has been acknowledged by the NRC staff in Generic Letter 84-15.

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.

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

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; also, Generic Letter 84-15, which modified the testing frequencies specified in Regulatory Guide 1.108.

Some of the Surveillance Requirements for demonstrating the operability of the diesel generators are modified by a footnote. The Specifications state the Surveillance Requirements are to be performed during shutdown, with the unit in mode 3 or higher. The footnote allows the particular surveillance to be performed during preplanned Preventative Maintenance (PM) activities that would result in the diesel generator being inoperable. The surveillance can be performed at that time as long as it does not increase the time the diesel generator is inoperable for the PM activity that is being performed. The footnote is only applicable at that time. The provision of the footnote shall not be utilized for operational convenience.

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."

In SURVEILLANCE 4.8.2.1.2.e, after the battery is returned to service (reconnected to and supplying its normal DC distribution center) following a performance discharge test (PDT), no discharge testing shall be done within 10 days on the other three batteries. This is a conservative measure to ensure the tested battery is fully charged. This restriction is an interim measure until the concern regarding recovered battery capacity immediately following recharging is resolved.

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-3 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-3 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Testing of these circuit breakers consists of injecting a current in excess of the breaker's nominal setpoint and measuring the response time. The measured response time is compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. 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.

Fuse testing is in accordance with IEEE Standard 242-1975. This program will detect any significant degradation of the fuses or improperly sized fuses. Safety is further assured by the "fail safe" nature of fuses, that is, if the fuse fails, the circuit will deenergize.

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 accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the minimum boron concentration value specified in the Core Operating Limits Report or greater includes a conservative uncertainty allowance of 50 ppm boron.

The Reactor Makeup Water Supply to the Chemical and Volume Control (NV) System is normally isolated during refueling to prevent diluting the Reactor Coolant System boron concentration. Isolation is normally accomplished by closing valve NV-250. However, isolation may be accomplished by closing valves NV-131, NV-140, NV-176, NV-468, NV-808, and either NV-132 or NV-1026, when it is necessary to makeup water to the Refueling Water Storage Tank during refueling operations.

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. In MODE 6 the Wide Range Neutron Flux Detectors (ENC) can be used as Source Range Neutron Flux Monitors. All of the LCO, ACTION, and SURVEILLANCE REQUIREMENTS of 3/4.9.2 must be met for the two Source Range Neutron Flux Monitors that are in use at any time.

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 accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY of the Reactor Building Containment Purge Exhaust System HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the Reactor Building Containment Purge Exhaust System HEPA filters and charcoal adsorbers and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis. The methyl iodide penetration test criteria for the carbon samples have been made more restrictive than required for the assumed iodine removal in the accident analysis because the humidity to be seen by the charcoal adsorbers may be greater than 70% under normal operating conditions.

BASES

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: (1) manipulator cranes will be used for movement of 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 POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident 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 RHR capability. With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

To prevent vortexing in the suction of the RHR pumps, the flow rate requirements for the RHR system were lowered from 3000 gpm to 1000 gpm. A specific surveillance has been added to ensure the flow remains high enough to ensure the reactor coolant system temperature remains less than or equal to 140 degrees-F. The problems associated with vortexing and mid-loop operations is provided in Generic Letter 88-17, Loss of Decay Heat.

3/4.9.9 and 3/4.9.10 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 FUEL HANDLING VENTILATION EXHAUST SYSTEM

The limitations on the Fuel Handling Ventilation Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing. The methyl iodide penetration test criteria for the carbon samples have been made more restrictive than required for the assumed iodine removal in the accident analysis because the humidity to be seen by the charcoal adsorbers may be greater than 70% under normal operating conditions.

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE

The requirements for spent fuel pool boron concentration specified in Specification 3.9.12 ensure that a minimum boron concentration is maintained in the pool. The requirements for spent fuel assembly storage specified in Specification 3.9.13 ensure that the pool remains subcritical. The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence the design of the spent fuel storage racks is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the spent fuel pool fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 4) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 1 or Region 2. This could increase the reactivity of the spent fuel pool. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water.

Tables 3.9-1 through 3.9-5 allow for specific criticality analyses for fuel which does not meet the requirements for storage defined in these tables. These analyses would require using NRC approved methodology to ensure that $k_{eff} \leq 0.95$ with a 95 percent probability at a 95 percent confidence level as described in Section 9.1 of the FSAR. This option is intended to be used for fuel not included in previous criticality analyses. Fuel storage is still limited to the configurations defined in TS 3.9-13. The use of specific analyses for qualification of previously unanalyzed fuel includes, but is not

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE (Continued)

limited to, fuel assembly designs not previously analyzed which may be as a result of new fuel designs or fuel shipments from another facility. Another more likely, and expected use of this specific analysis provision would be to analyze movement and storage of individual fuel pins as a result of reconstitution activities.

In verifying the design criteria of $k_{eff} \leq 0.95$, the criticality analysis assumed the most conservative conditions, i.e. fuel of the maximum permissible reactivity for a given configuration. Since the data presented in Specification 3.9.13.a and 3.9.13.b represents the maximum reactivity requirements for acceptable storage, substitutions of less reactive components would also meet the $k_{eff} \leq 0.95$ criteria. Hence, any non-fuel component may be placed in a designated empty cell location. Likewise, an empty cell, or a non-fuel component may be substituted for any designated fuel assembly location. These, or other substitutions which will decrease the reactivity of a particular storage cell will only decrease the overall reactivity of the spent fuel storage pool.

If both restricted and unrestricted storage is used in Region 1, an additional criteria has been imposed to ensure that the boundary row between these two configurations would not locally increase the reactivity above the required limit. Likewise if checkerboard storage is used in Region 2, an additional restriction has been imposed on the boundaries of the checkerboard storage region to ensure that the reactivity would not increase above the required limit. No other restrictions on region interfaces are necessary.

For storage in Region 2 requiring loading pattern restrictions, (per Specifications 3.9.13.b.2 or 3.9.13.b.3) fuel may be stored in either the "cell" or "non-cell" locations. "Cell" locations are the areas inside the fabricated storage cells and "non-cell" locations are the storage locations created by arranging the fabricated storage cells in a checkerboard configuration. Hence the "non-cell" locations are the areas defined by the outside walls of the 4 adjacent "cell" locations.

The action statement applicable to fuel storage in the spent fuel pool requires that action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. Prior to the resumption of fuel movement, the requirements of the LCOs must be met. This requires restoring the soluble boron concentration and the correct fuel storage configuration to within the corresponding limits. This does not preclude movement of a fuel assembly to a safe position.

The surveillance requirements ensure that the requirements of the two LCOs are satisfied, namely boron concentration and fuel placement. The boron concentration in the spent fuel pool is verified to be greater than or equal to the minimum limit. The fuel assemblies are verified to meet the subcriticality requirement by meeting either the initial enrichment and burnup

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE (Continued)

requirements of Table 3.9-1 through 3.9-5, or by using NRC approved methodology to ensure that $k_{eff} \leq 0.95$. By meeting either of these requirements, the analyzed accidents are fully addressed.

The fuel storage requirements and restrictions discussed here and applied in section 3.9.13 are based on a maximum allowable fuel enrichment of 4.75 weight% U-235. The enrichments listed in Tables 3.9-1 through 3.9-5 are nominal enrichments and include uncertainties to account for the tolerance on the as built enrichment. Hence the as built enrichments may exceed the enrichments listed in the tables by up to 0.05 weight% U-235. Qualifying burnups for enrichments not listed in the tables may be linearly interpolated between the enrichments provided. This is because the reactivity of an assembly varies linearly for small ranges of enrichment.

REFERENCES

1. "Regulatory Guide 1.13: Spent Fuel Storage Facility Design Basis", U.S. Nuclear Regulatory Commission, Office of Standards Development, Revision 1, December 1976.

2. "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations", American Nuclear Society, ANSI N210-1976/ANS-57.2, April 1976.

3. FSAR, Section 9.1.

4. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

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 NOT USED

3/4.11.1.2 NOT USED

3/4.11.1.3 NOT USED

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 NOT USED

3/4.11.2.2 NOT USED

3/4.11.2.3 NOT USED

3/4.11.2.4 NOT USED

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a Waste Gas System leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the WASTE GAS HOLDUP SYSTEM to assure that a release would be substantially below the dose guideline values of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

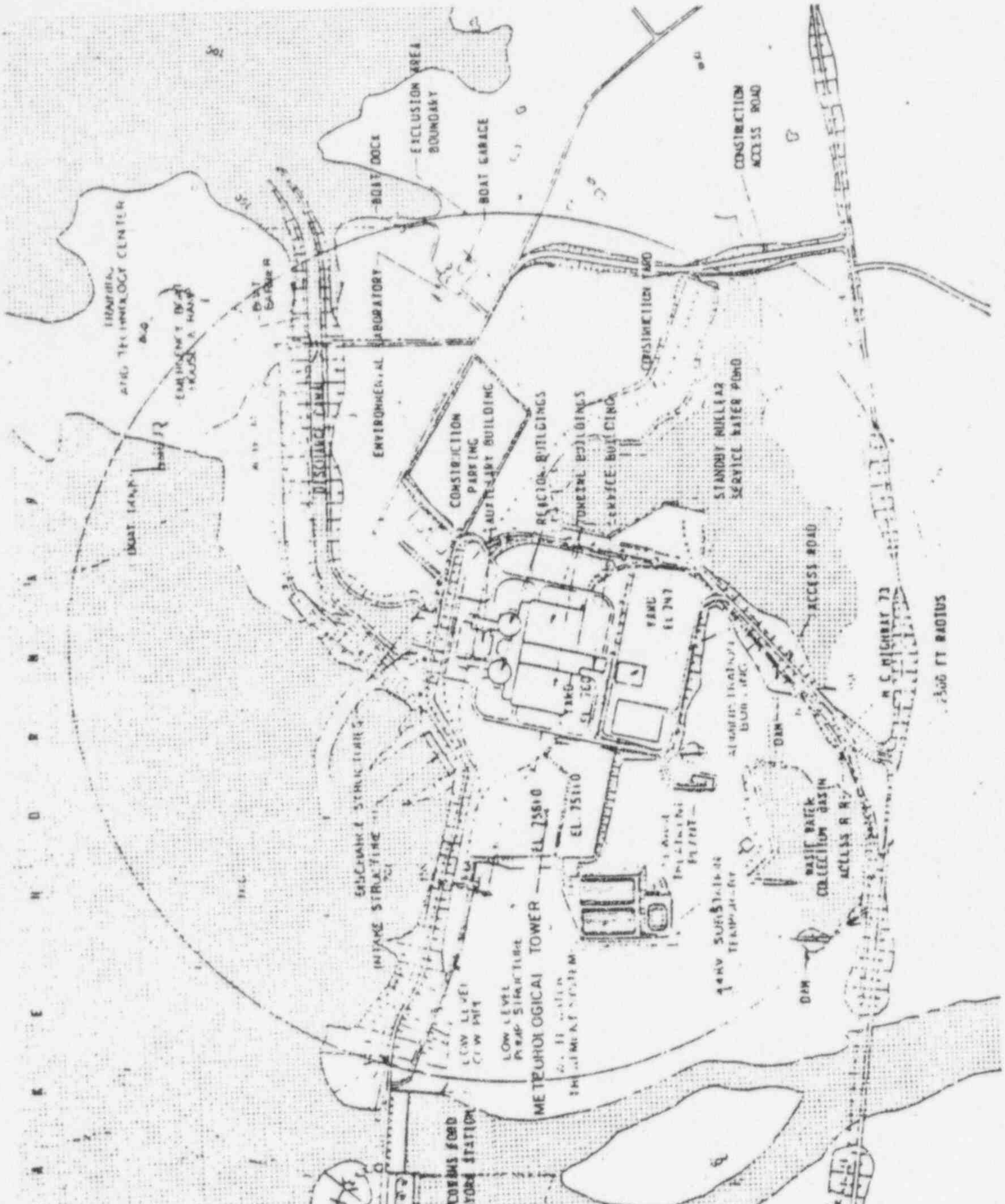
5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

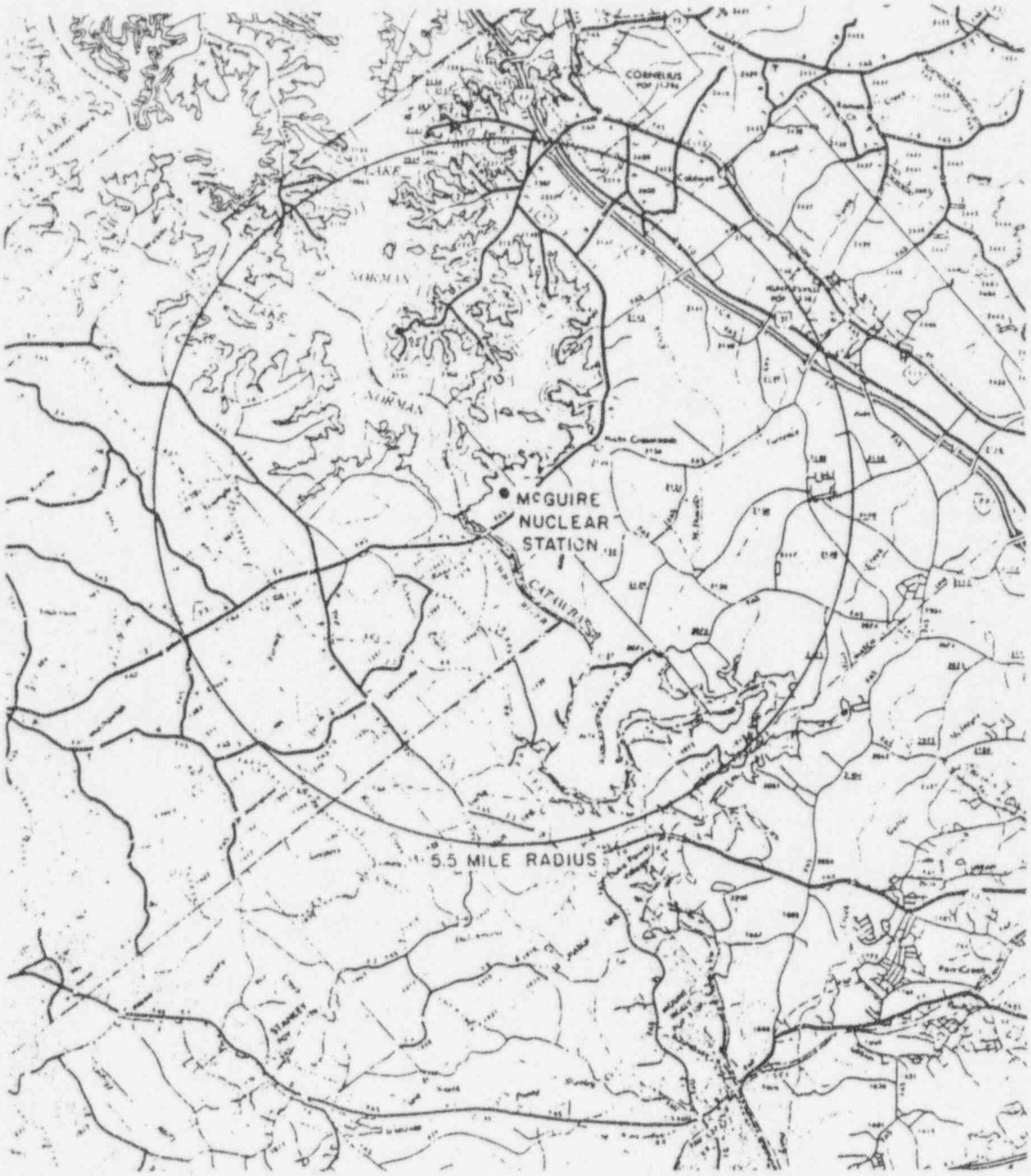
MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4. The definition of UNRESTRICTED AREA used in implementing the Radiological Effluent Technical Specifications has been expanded over that in 10 CFR Part 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR Part 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR Part 50.36a.



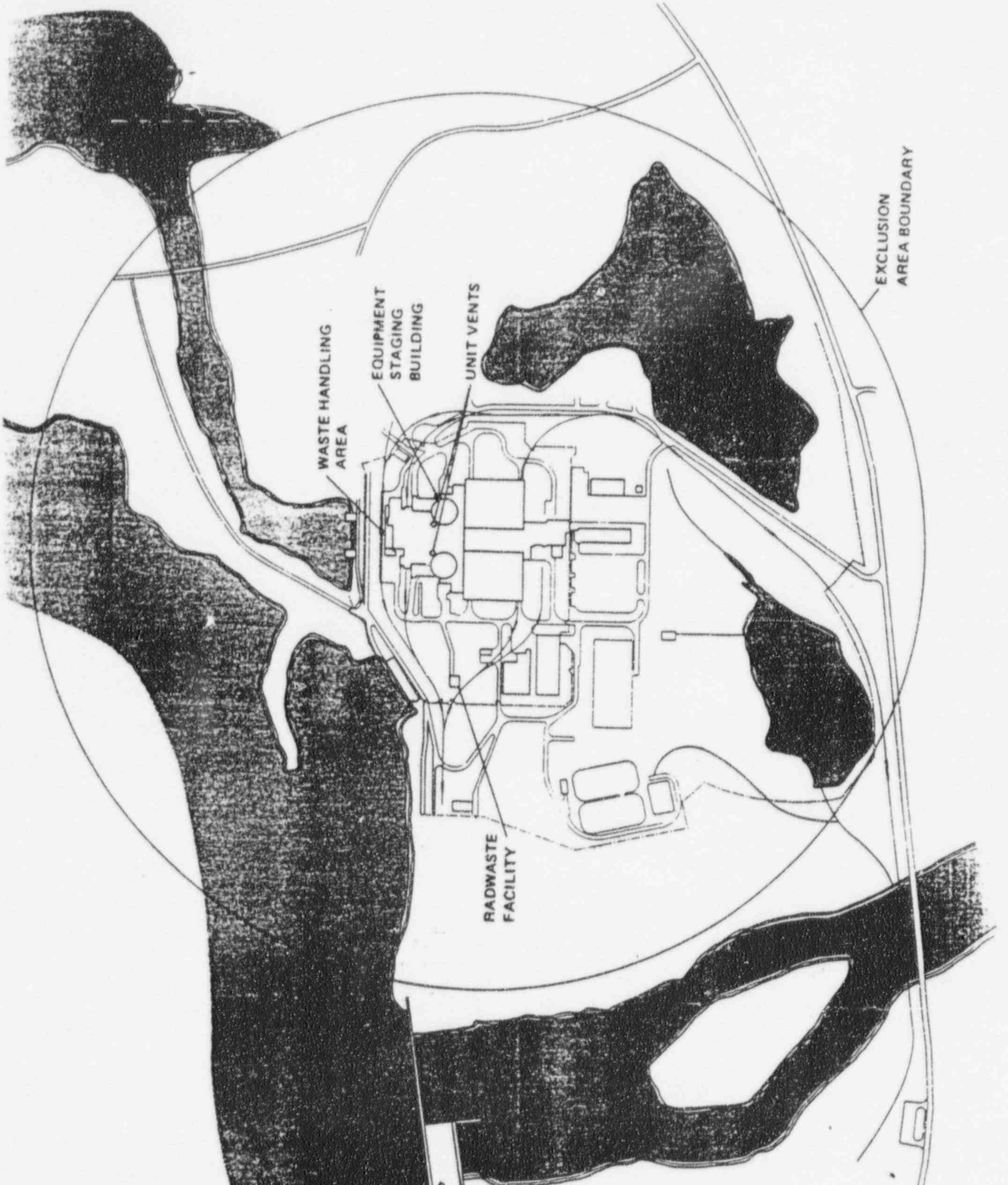
EXCLUSION AREA
McGUIRE NUCLEAR STATION

FIGURE 5.1-1



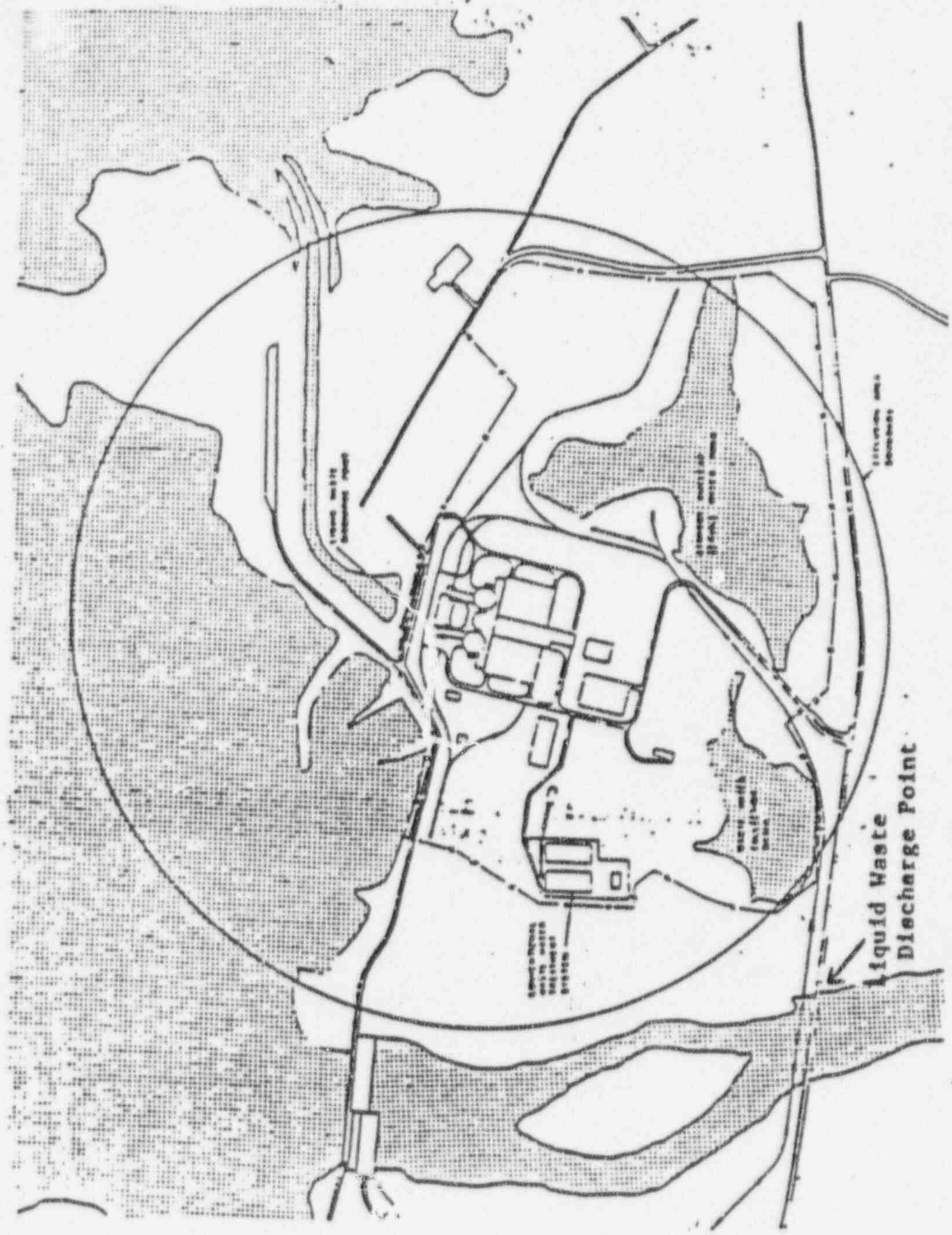
LOW POPULATION ZONE
McGUIRE NUCLEAR STATION

FIGURE 5.1-2



SITE BOUNDARY FOR
GASEOUS EFFLUENTS
McGUIRE NUCLEAR STATION

FIGURE 5.1-3



SITE BOUNDARY FOR
LIQUID EFFLUENTS
McGUIRE NUCLEAR STATION

FIGURE 5.1-4

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel surrounded by a concrete containment.

5.2.1.1 CONTAINMENT VESSEL

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 169 feet.

c. Net free volume = 1.2×10^6 cubic feet.

d. Nominal thickness of vessel walls = 0.75 inch.

e. Nominal thickness of vessel dome = 0.6875 inch.

f. Nominal thickness of vessel bottom = 0.25 inch.

5.2.1.2 REACTOR BUILDING

a. Nominal annular space = 5 feet.

b. Annulus nominal volume = 427,000 cubic feet.

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 177 feet.

d. Nominal inside diameter = 125 feet.

e. Cylinder wall minimum thickness = 3 feet.

f. Dome minimum thickness = 2.25 feet.

g. Dome inside radius = 87 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 15.0 psig and a temperature of 250°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses

FUEL ASSEMBLIES (Continued)

to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-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 for Unit 2 control rods shall be 100% boron carbide (B_4C) for 102 inches and 80% silver, 15% indium, and 5% cadmium for the 40-inch tip. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,040 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) A nominal 10.4" center to center distance between fuel assemblies placed in Region 1; and

3) A nominal 9.125" center to center distance between fuel assemblies placed in Region 2.

b. The new fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) $k_{eff} \leq 0.98$ if moderated by aqueous foam as described in Section 9.1 of the FSAR; and

3) A nominal 21" center to center distance between fuel assemblies placed in the storage racks.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745 ft. 7 in.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1463 fuel assemblies (286 spaces in Region 1 and 1177 spaces in Region 2).

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/hr and 200 cooldown cycles at ≤ 100°F/hr (pressurizer cooldown at ≤ 200°F/hr for 200°F ≤ T pressurizer ≤ 650°F).	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 551°F. Cooldown cycle - T_{avg} from ≥ 551°F to ≤ 200°F.
	80 loss of load cycles.	Without immediate Turbine or Reactor trip.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical power source supplying the Onsite Class 1E Distribution 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.
	200 large step decreases in load.	100% to 0% of RATED THERMAL POWER with steam dump.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	1 main reactor coolant pipe break.	Break in a reactor coolant pipe > 6 inches equivalent diameter.
	Operating Basis Earthquakes.	400 cycles - 20 earthquakes of 20 cycles each.
	50 leak tests.	Pressurized to 2500 psig.
	5 hydrostatic pressure tests.	Pressurized to 3107 psig.
Reactor Vessel	Operating Basis Earthquakes.	50 cycles.
Secondary Coolant System	1 steam line break.	Break in a steam line ≥ 6.0 inches equivalent diameter.
	5 hydrostatic pressure tests.	Pressurized to 1481 psig.

SECTION 6.0
ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice-President McGuire Nuclear Site shall be reissued to all McGuire Nuclear Site personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationship, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.

b. The Station Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President of McGuire Nuclear Site shall have responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The Senior Vice President Nuclear Generation Department will be the Senior Nuclear Executive and have corporate responsibility for overall nuclear safety.

e. The individuals who train the operating staff and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

6.2.2 The unit organization shall be as shown in the FSAR, Chapter 13, and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator for each unit shall be in the control room when fuel is in either reactor. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. A Radiation Protection Technician shall be on site when fuel is in either reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

e. Administrative procedures shall be developed and implemented to limit the working hours of station staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, radiation protection technicians, non-licensed operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 12 hour day with alternating 48 hour and 36 hour work week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 28 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time;

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time; and

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Station Manager or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Station Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3 or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
SS	1	1	1
SRO	1	none[b]	1
RO	3[a]	2[a]	3[a]
AO	3[a]	3[a]	3[a]
SM	1	none	1

SS - Shift Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary operator
SM - Shift Manager

[a] At least one of the required individuals must be assigned to the designated position for each unit.
[b] At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Manager*) 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 or Operator license shall be designated to assume the control room command function.

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*On occasions when there is a need for both the Shift Supervisor and the SRO to be absent from the control room, the Shift Manager shall be allowed to assume the control room command function and serve as the SRO in the control room provided that: (1) the Shift Supervisor is available to return to the control room within 10 minutes, (2) the assumption of SRO duties by the Shift Manager be limited to periods not in excess of 15 minutes duration and a total time not to exceed 1 hour during any 8-hour shift, and (3) the Shift Manager has an SRO license on the unit.

6.2.3 McGUIRE SAFETY REVIEW GROUP

FUNCTION

6.2.3.1 The McGuire Safety Review Group (SRG) shall function to provide the review of plant design and operating experience for potential opportunities to improve plant safety; evaluation of plant operations and maintenance activities; and, to advise management on the overall quality and safety of plant operations. The SRG shall make recommendations for revised procedures, equipment modifications, or other means of improving plant safety to appropriate station/corporate management.

COMPOSITION

6.2.3.2 The SRG shall be composed of at least five individuals and at least three of these shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his/her field, at least 1 year of which experience shall be in the nuclear field.

The remaining individuals in the SRG shall have either (1) at least 5 years of nuclear experience and hold or have held a Senior Reactor Operator license; or (2) at least 8 years of professional level experience in his/her field, at least 5 years of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The SRG shall be responsible for:

a. Review of selected plant operating characteristics and other appropriate sources of plant design and operating experience information for awareness and incorporation into the performance of other duties.

b. Review of the effectiveness of corrective actions taken as a result of the evaluation of selected plant operating characteristics and other appropriate sources of plant design and operating experience information.

c. Review of selected programs, procedures, and plant activities, including maintenance, modification, operational problems, and operational analysis.

d. Surveillance of selected plant operations and maintenance activities to provide independent verification* that they are performed correctly and that human errors are reduced to as low as practicable.

e. Investigation of selected unusual events and other occurrences as assigned by Station Management or the Manager of Safety Assurance.

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*Not responsible for sign-off function.

6.2.3 McGUIRE SAFETY REVIEW GROUP (Continued)

AUTHORITY

6.2.3.4 The SRG shall report to and advise the Manager of Safety Assurance, on those areas of responsibility specified in Section 6.2.3.

RECORDS

6.2.3.5 Records of activities performed by the SRG shall be prepared and maintained for the life of the station. Summary reports of activities performed by the SRG shall be forwarded each calendar month to the Manager of Safety Assurance.

6.2.4 SHIFT MANAGER

6.2.4.1 The Shift Manager, whose functions include those of a Shift Technical Advisor, shall serve in an advisory capacity to the Shift Supervisor.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the SRG.

6.5 REVIEW AND AUDIT

6.5.1 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

Programs shall be established for the preparation, review, approval, and retention of documents required by the activities described in Specifications 6.5.1.1 through 6.5.1.11. Approvals shall be by the head of the appropriate site organization, the head of the appropriate station organization, the head of the appropriate site engineering organization, the head of the environmental organization, or an alternate as specified in other applicable regulatory documents or administrative controls.

6.5.1.1 Each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, shall be prepared by a knowledgeable individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/organization other than the individual/organization which prepared the procedure, or changes thereto. Procedures, or changes thereto, shall be approved in accordance with Specifications 6.8.2 and 6.8.3.

6.5.1.2 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a knowledgeable individual/organization. Each such modification shall be reviewed by an individual/organization other than the individual/organization which designed the modification.

6.5.1.3 Individuals responsible for reviews performed in accordance with Specifications 6.5.1.1 and 6.5.1.2 shall be members of the supervisory staff assigned to the site, previously designated by the Site Vice President to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated site review personnel.

6.5.1.4 Proposed changes to the Appendix A Technical Specifications shall be prepared by a knowledgeable individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/organization other than the individual/organization which prepared the proposed change.

6.5.1.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be prepared and approved in a manner identical to that of Specification 6.5.1.1. These proposed tests and experiments shall be reviewed by a knowledgeable individual/organization other than the individual/organization which prepared the proposed tests and experiments.

ACTIVITIES (Continued)

6.5.1.6 ALL REPORTABLE EVENTS and all violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be reviewed by a knowledgeable individual/organization other than the individual/organization which prepared the report.

6.5.1.7 Special reviews and investigations, and the preparation of reports thereon, shall be performed by a knowledgeable individual/organization.

6.5.1.8 A knowledgeable individual/organization shall review every unplanned onsite release of radioactive material to the environs and prepare reports covering evaluation, recommendations, and disposition of the corrective ACTION to prevent recurrence.

6.5.1.9 A knowledgeable individual/organization shall review changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and Radwaste Treatment Systems.

6.5.1.10 A knowledgeable individual/organization shall review the Fire Protection Program and implementing procedures.

6.5.1.11 Reports documenting each of the activities performed under Specifications 6.5.1.1 through 6.5.1.10 shall be maintained.

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB)

FUNCTION

6.5.2.1 The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

FUNCTION (Continued)

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Administrative control and quality assurance practices.

ORGANIZATION

6.5.2.2 The Director, members and alternate members of the NSRB shall be appointed in writing by the Senior Vice President, Nuclear Generation and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of 5 years technical experience, of which a minimum of 3 years shall be in one or more areas given in Specification 6.5.2.1. In special cases, candidates for appointment without an academic degree in engineering or physical science may be qualified with a minimum of ten years experience in one of the areas in Specification 6.5.2.1. No more than two alternates shall participate as voting members in NSRB activities at any one time.

6.5.2.3 The NSRB shall be composed of at least five members, including the Director. Members of the NSRB may be from the Nuclear Generation Department, from other departments within the Company, or from external to the Company. A maximum of one member of the NSRB may be from the McGuire Nuclear Site staff.

6.5.2.4 Consultants shall be utilized as determined by the NSRB Director to provide expert advice to the NSRB.

6.5.2.5 Staff assistance may be provided to the NSRB in order to promote the proper, timely, and expeditious performance of its functions.

6.5.2.6 The NSRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least twice per year thereafter.

6.5.2.7 The quorum of the NSRB necessary for the performance of the NSRB review and audit functions of these Technical Specifications shall consist of the Director, or designated alternate, and at least four other NSRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of McGuire Nuclear Station.

REVIEW

6.5.2.8 The NSRB shall review:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

d. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

e. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

f. All REPORTABLE EVENTS;

g. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems or components that could affect nuclear safety;

h. Quality Assurance Program audits relating to station operations and actions taken in response to these audits; and

i. Reports of activities performed under the provisions of Specifications 6.5.1.1 through 6.5.1.10.

AUDITS

6.5.2.9 Audits of site activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The performance, training, and qualifications of the entire station staff;

AUDITS (Continued)

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50;

e. The Emergency Plan and implementing procedures;

f. The Security Plan and implementing procedures;

g. The Facility Fire Protection programmatic controls including the implementing procedures;

h. The fire protection equipment and program implementation utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

i. The Radiological Environmental Monitoring Program and the results thereof;

j. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures;

k. The PROCESS CONTROL PROGRAM and implementing procedures for SOLIDIFICATION of radioactive wastes;

l. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring and;

m. Any other area of site operation considered appropriate by the NSRB or the Senior Vice President, Nuclear Generation.

AUTHORITY

6.5.2.10 The NSRB shall report to and advise the Senior Vice President, Nuclear Generation on those areas of responsibility specified in Specifications 6.5.2.8 and 6.5.2.9.

RECORDS

6.5.2.11 Records of NSRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRB meeting shall be prepared, approved, and forwarded to the Senior Vice President, Nuclear Generation and to the Site Vice President, within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.8 above, shall be prepared, approved and forwarded to the Senior Vice President, Nuclear Generation, and to the Site Vice President, within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.9 above, shall be forwarded to the Senior Vice President, Nuclear Generation and to the Site Vice President, and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Station Manager; or for the Station Manager by: (1) the Operations Superintendent, (2) the Maintenance Superintendents, (3) the Work Control Superintendent, as previously designated by the Station Manager, and the results of the review shall be submitted to the NSRB and the Site Vice President.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice President, and the NSRB shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Operations Superintendent and the Station Manager. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence;

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Site Vice President, within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The applicable procedures required to implement the requirements of NUREG-0737;

c. Deleted

d. Deleted

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation; and

g. Quality Assurance Program for effluent and environmental monitoring.

h. Technical Review and Control Program implementation.

i. Fire Protection Program implementation.

j. Plant Operations Review Committee implementation.

k. Commitments contained in FSAR Chapter 16.0

6.8.2 Each procedure of Specification 6.8.1 above, and changes thereto, shall be reviewed and approved by an appropriate division manager, superintendent/manager, or one of their designated direct reports prior to implementation and shall be reviewed periodically as set forth in administrative procedures. For procedures which implement offsite environmental, technical, and laboratory activities, the above review and approval may be performed by the General Manager, Environmental Services or designee.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above 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 approved by an appropriate division manager, superintendent/manager, or one of their designated direct reports within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Reactor 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 RHR, Boron Recycle, Refueling Water, Liquid Waste, Waste Gas, Safety Injection, Chemical and Volume Control, Containment Spray, and Nuclear Sampling. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) 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,

PROCEDURES AND PROGRAMS (Continued)

4) Procedures for the recording and management of data,

5) Procedures defining corrective actions for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1) Training of personnel, and

2) Procedures for monitoring.

e. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines, and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

f. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in FSAR Chapter 16, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and set-point determination in accordance with the methodology in the Offsite Dose Calculation Manual (ODCM),

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 times 10 CFR Part 20.1001-20.2401, Appendix B, Table 2, Column 2,

PROCEDURES AND PROGRAMS (Continued)

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following:

   a. For noble gases: Less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

   b. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrems/yr to any organ.

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

PROCEDURES AND PROGRAMS (Continued)

g. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in FSAR Chapter 16, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant STARTUP and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the License involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

STARTUP REPORT (Continued)

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in License conditions based on other commitments shall be included in this report.

6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the STARTUP test program, or (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 [1/]

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Annual Reports shall include the activities of the unit as described below:

a. Personnel Exposures

Reports required on an annual basis shall include 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, 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.

[1/] 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[1/]

b. Primary Coolant Specific Activity

Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be include: 1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; 2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded 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 system flow history starting 48 hours prior to the first sample in which the limit was exceeded; 4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and 5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 Routine 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 report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

1/ 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.

* A single submittal may be made for a multiple unit station.

** 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.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference limits, target band*, and APL^{ND}* for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, F^{RTP}_{Q}, K(Z), W(Z)**, APL^{ND}** and $W(Z)_{BL}$** for Specification 3/4.2.2, and

6. Nuclear Enthalpy Rise Hot Channel Factor, $F^{L}_{\Delta H}$***, or $F^{RTP}_{\Delta H}$****, and Power Factor Multiplier, $MF_{\Delta H}$****, limits for Specification 3/4.2.3.

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 2.2.1.

8. Boric Acid Storage System and Refueling Water Storage Tank volume and boron concentration limits for Specifications 3/4.1.2.5 and 3/4.1.2.6.

9. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3/4.5.1 and 3/4.5.5.

10. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

11. Spent fuel pool boron concentration limits for Specification 3/4.9.12.

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* Reference 5 is not applicable to target band and APL^{ND}.

** References 4 and 5 are not applicable to W(Z), APL^{ND}, and $W(Z)_{BL}$.

*** Reference 1 is not applicable to $F^{L}_{\Delta H}$.

**** Reference 5 is not applicable to $F^{RTP}_{\Delta H}$ and $MF_{\Delta H}$.

CORE OPERATING LIMITS REPORT

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987 (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," SER dated January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

5. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990 (DPC Proprietary).

   (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NE-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3/4.9.12 - Spent Fuel Pool Boron Concentration.)

CORE OPERATING LIMITS REPORT

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, "Thermal-Hydraulic Transient Analysis Methodology," August 1994.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November, 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H(X,Y)$.)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation setpoints).

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

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.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

RECORD RETENTION (Continued)

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 reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

j. Records of meetings of the NSRB and reports required by Specification 6.5.1.11;

k. Records of the service lives of all snubbers including the date at which the service life commences and associated installation and maintenance records;

l. Records of secondary water sampling and water quality; and

m. 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.

n. Records of reviews performed for changes made to the ODCM and the PCP.

6.10.3 Records of quality assurance activities required by the QA Manual shall be retained for a period of time required by ANSI N45.2.9-1974.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mrem/hr at 45 CM (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Radiation Protection Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mrem/hr provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

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 level in the area has 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 Radiation Protection Manager 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 mrem/hr 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 area 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 areas accessible to personnel with radiation levels greater than 1000 mrem/hr* that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, that area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2n. This documentation shall contain:

*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (PCP) (Continued)

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective upon review and acceptance by the Station Manager and a qualified individual/organization.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2n. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective upon review and acceptance by the station manager and a qualified individual/organization.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee initiated major changes to the Radioactive Waste Systems (liquid, gaseous and solid)

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Station Manager. The discussion of each change shall contain:

   1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

   2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

   4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

   5) An evaluation of the change, which shows expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

   6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

   7) An estimate of the exposure to plant operating personnel as a result of the change; and

   8) Documentation of the fact that the change was reviewed and found acceptable by the Station Manager or the Chemistry Manager.

b. Shall become effective upon review and acceptance by a qualified individual/organization.

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*Licensees may choose to submit the information called for in this specification as part of the annual FSAR update.

TECHNICAL SPECIFICATIONS

FOR

McGUIRE NUCLEAR STATION

UNIT NO. 1

DOCKET NO. 50-369

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

DEFINITIONS

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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 with 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 operator action during periods when containment isolation valves may be opened under administrative controls pursuant to Specification 4.6.1.1.a; or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions.

b. All equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

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

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

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

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

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

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

MASTER RELAY TEST

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

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by FSAR Chapter 16.

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 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.

PROCESS CONTROL PROGRAM (PCP)

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

PURGE - PURGING

1.24 PURGE or PURGING 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 required to purify the confinement.

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

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

REACTOR BUILDING INTEGRITY

1.27 REACTOR BUILDING INTEGRITY shall exist when:

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

b. The Annulus Ventilation System is in compliance with the requirements of Specification 3.6.1.8, and

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

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 to 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.

SOLIDIFICATION

1.33 Not Used

SOURCE CHECK

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

STAGGERED TEST BASIS

1.35 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

WASTE GAS HOLDUP SYSTEM

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

TABLE 1.1

FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL MODES

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

*Excluding decay heat.

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

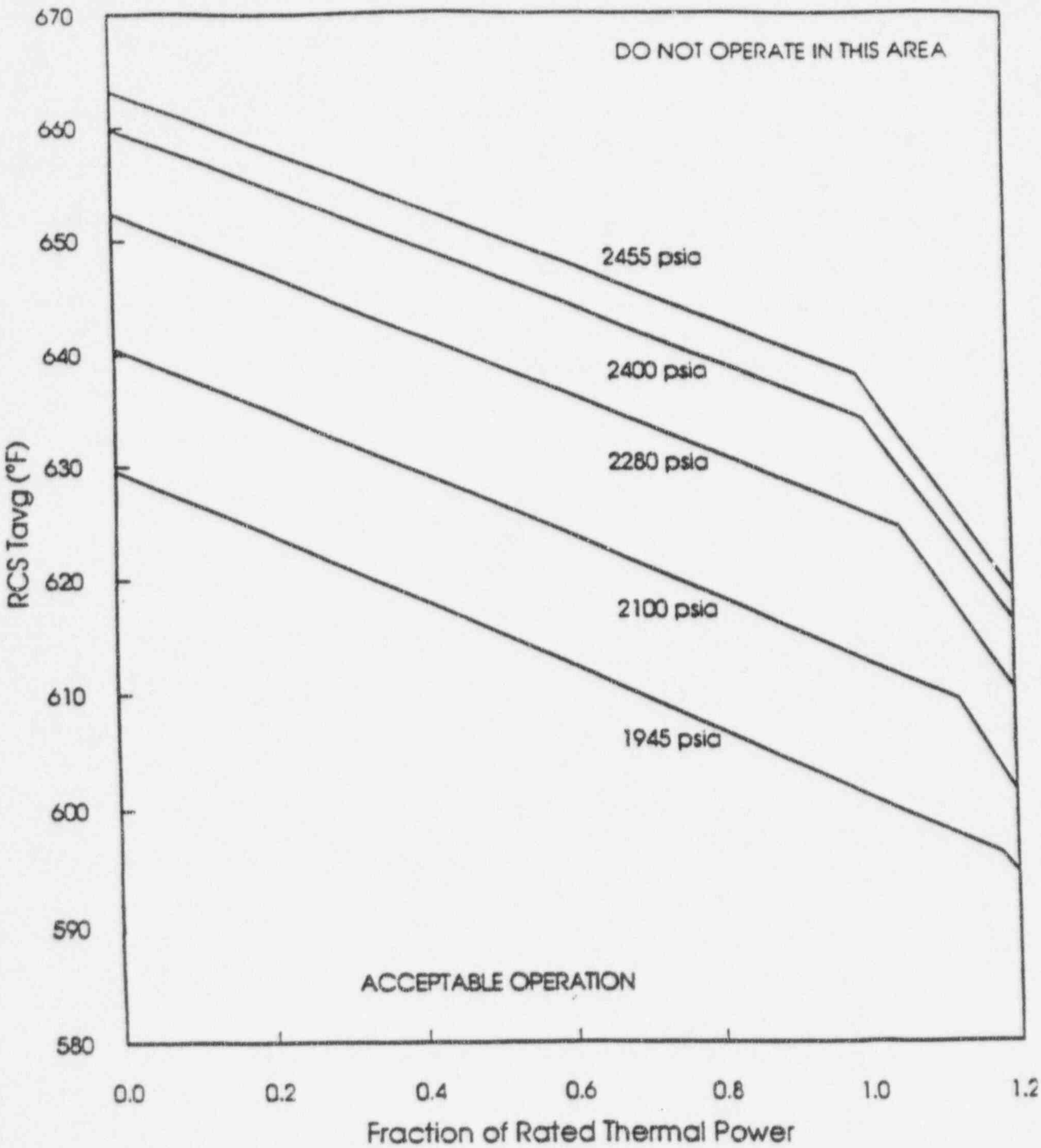

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS -
FOUR LOOPS IN OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
5. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
6. Overtemperature ΔT	See Note 1	See Note 3
7. Overpower ΔT	See Note 2	See Note 4
8. Pressurizer Pressure--Low	$\geq$ 1945 psig	$\geq$ 1935 psig
9. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
10. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
11. Low Reactor Coolant Flow	$\geq$ 91% of minimum measured flow per loop*	$\geq$ 90% of minimum measured flow per loop*

*Minimum measured flow is 95,500 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
12. Steam Generator Water Level--Low-Low	≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40% of span at 100% of RATED THERMAL POWER	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing to 39.0% of span at 100% of RATE THERMAL POWER.
13. Undervoltage-Reactor Coolant Pumps	≥ 5082 volts-each bus	≥ 5016 volts-each bus
14. Underfrequency-Reactor Coolant Pumps	≥ 56.4 Hz - each bus	≥ 55.9 Hz - each bus
15. Turbine Trip		
a. Low Trip System Pressure	≥ 45 psig	≥ 42 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	≥ 1 x 10^{-10} amps	≥ 6 x 10^{-11} amps

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
2) P-13 Input	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8, Low Reactor Coolant Loop Flow, and Reactor Coolant Pump Breaker Position	≤ 48% of RATED THERMAL POWER	≤ 49% of RATED THERMAL POWER
d. Low Setpoint Power Range Neutron Flux, P-10, Enable Block of Source Intermediate and Power Range Reactor Trips	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
e. Turbine Impulse Chamber Pressure, P-13, Input to Low Power Reactor Trips Block P-7	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent
18. Reactor Trip Breakers	N.A.	N.A.
19. Automatic Trip and Interlock Logic	N.A.	N.A.

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

NOTE 1 OVERTEMPERATURE ΔT

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

Where:

ΔT = Measured ΔT by Loop Narrow Range RTD,

ΔT_0 = Indicated ΔT at RATED THERMAL POWER,

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

τ_1, τ_2 = Time constants utilized in the lead-lag controller for ΔT, as presented in the Core Operating Limits Report,

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

τ_3 = Time constant utilized in the lag compensator for ΔT, as presented in the Core Operating Limits Report,

K_1 = Overtemperature ΔT reactor trip setpoint as presented in the Core Operating Limits Report,

K_2 = Overtemperature ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report,

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

τ_4, τ_5 = Time constants utilized in the lead-lag controller for T_{avg}, as presented in the Core Operating Limits Report,

T = Average temperature, °F,

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

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

NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report,

T' = $\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report,

P = Pressurizer pressure, psig,

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

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

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

(i) for $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the Core Operating Limits Report; $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

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

NOTE 2: OVERPOWER ΔT

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

Where:

ΔT	=	As defined in Note 1,
ΔT_0	=	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,
K_4	=	Overpower ΔT reactor trip setpoint as presented in the Core Operating Limits Report,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation,
τ_7	=	Time constant utilized in the rate-lag controller for T_{avg}, as presented in the Core Operating Limits Report,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,
K_6	=	Overpower ΔT reactor trip heatup setpoint penalty coefficient as presented in the Core Operating Limits Report for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

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

T = As defined in Note 1,

T″ = $\leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER,

S = As defined in Note 1, and

f_2 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between the "positive" and "negative" f_2 (ΔI) breakpoints as presented in the Core Operating Limits Report; f_2 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and q_t + q_b is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of q_t - q_b is more negative than the f_2 (ΔI) "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of q_t - q_b is more positive than the f_2 (ΔI) "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the f_2 (ΔI) "positive" slope presented in the Core Operating Limits Report.

NOTE 3: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.4% of Rated Thermal Power.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% of Rated Thermal Power.

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. This relation 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), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

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

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

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

The curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.50 and a reference cosine axial power shape with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.50\ [1 + (1/RRH)\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER, and
RRH is given in the COLR.

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The 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 settings is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the accident analysis to enhance the overall reliability of the Reactor Trip System.

The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance for all trips including those trips assumed in the safety analyses.

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.

Power Range, Neutron Flux (Continued)

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 Positive Rate

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

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.

Overtemperature ΔT

The Overtemperature Delta T 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 response time delays associated with the RTDs mounted in thermowells, 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.

Overpower ΔT

The Overpower Delta T trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under overpower conditions, limits the required range for overtemperature delta T protection, 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, (2) rate of change of temperature for dynamic compensation for instrumentation delays associated with the loop temperature detectors, and (3) axial power distribution, 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 Break."

Pressurizer Pressure

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

Low Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide 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 91% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 91% of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on loss of flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

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

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.5 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.6 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Turbine trip is automatically blocked by P-8 (a power level of approximately 48% of RATED THERMAL POWER with a turbine impulse chamber at approximately 48% of full power equivalent); and on increasing power, reinstated automatically by P-8.

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 Reactor trip and de-energizing of the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

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

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

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

P-13 Provides input to P-7.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

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

a. At least HOT STANDBY within the next 6 hours,
b. At least HOT SHUTDOWN within the following 6 hours, and
c. At least COLD SHUTDOWN within the subsequent 24 hours.

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

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. 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 maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% 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 1.0 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

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

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of 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 Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

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

1) Reactor coolant system boron concentration,

2) Control rod position,

3) Reactor coolant system average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% 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 within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0.

APPLICABILITY: Figure 3.1-0 and COLR Figure 1 Limits - MODES 1 and 2* only.#
End of Cycle Life (EOL) Limit - MODES 1, 2, and 3 only.#

ACTION:

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

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

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

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

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

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

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

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

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

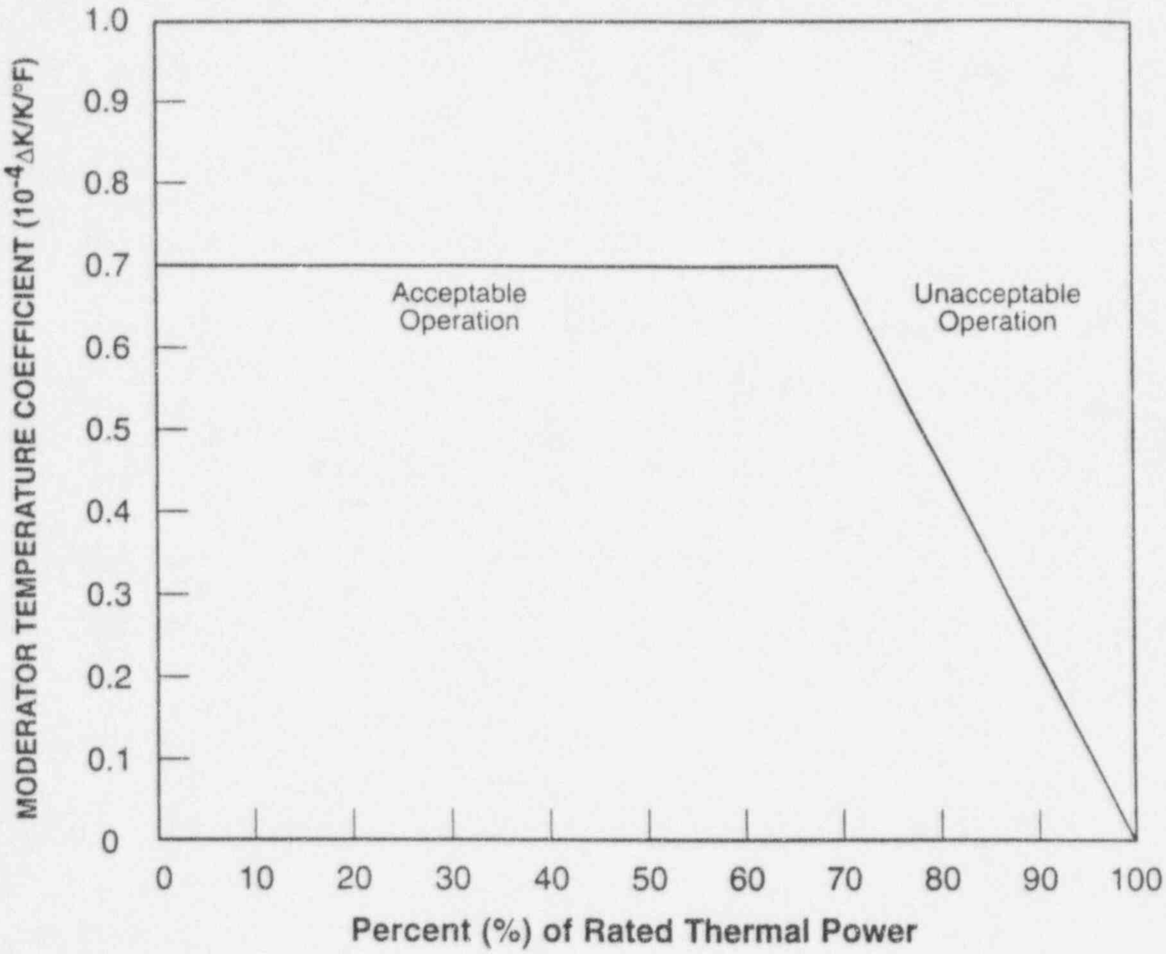

FIGURE 3.1-0
MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2#*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

#With K_{eff} greater than or equal to 1.0.

*See Special Test Exception 3.10.3.

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

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°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 a differential pressure across the pump of greater than or equal to 2380 psid is developed when tested pursuant to Specification 4.0.5.

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

#Two charging pumps may be operable and operating for $\leq$15 minutes to allow swapping charging pumps.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two# charging pumps shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.1.2.4.2 All centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each charging pump has been isolated from the Reactor Coolant System by at least two isolation valves with power removed from the valve operators.

#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. Two charging pumps may be operable and operating for ≤15 minutes to allow swapping charging pumps.

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 and at least one associated Heat Tracing System with:

1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

2) A minimum boron concentration as presented in the Core Operating Limits Report, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

2) A minimum boron concentration as presented in the Core Operating Limits Report, and

3) A minimum solution temperature of 70°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

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

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System and at least one associated Heat Tracing System with:

   1) A minimum contained borated water volume as presented in the Core Operating Limits Report,

   2) A minimum boron concentration as presented in the Core Operating Limits Report, and

   3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

   1) A minimum contained borated water volume as presented in the Core Operating Limits Report or Specification 3.5.5a, whichever is larger,

   2) A minimum boron concentration as presented in the Core Operating Limits Report,

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

   4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

REACTOR COOLANT SYSTEM

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 primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

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

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

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

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

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

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

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

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

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

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

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

    2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. 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;

*See Special Test Exceptions 3.10.2 and 3.10.3.

ACTION (Continued)

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

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

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

d. With more than one full-length 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 Specification 3.1.3.6. 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.

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 Misoperation

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

Major Secondary System Pipe Rupture

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

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 18 months. The Reactor Trip System Breakers can be closed in order to perform this surveillance.

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

#See Special Test Exception 3.10.5.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.2 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 (*) of RATED THERMAL POWER.

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.

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

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 12 hours thereafter.

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*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.0.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits specified in the COLR, 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 insertion limits specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

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

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*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.0.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the acceptable limits as specified in the Core Operating Limits Report (COLR).

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

ACTION:

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

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

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

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   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 Monitoring Alarm to OPERABLE status.

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

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

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*See Special Test Exception 3.10.2.

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(X,Y,Z)$ shall be limited by imposing the following relationship:

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

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

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

$K(Z)$ = the normalized $F_Q(X,Y,Z)$ limit specified in the COLR for the appropriate fuel type, and

$F_Q^{MA}(X,Y,Z)$ = the measured heat flux hot channel factor $F_Q^M(X,Y,Z)$ with the adjustments specified in 4.2.2.3

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(X,Y,Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours, and

b. Control the AFD to within new AFD limits which are determined by reducing the allowable power at each point along the AFD limit lines of Specification 3.2.1 at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and reset the AFD alarm setpoints to the modified limits within 8 hours, and

c. POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit, and

d. 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 $F_Q(X,Y,Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q{}^M(X,Y,Z)$[1] shall be evaluated to determine whether $F_Q(X,Y,Z)$ 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. Measuring $F_Q{}^M(X,Y,Z)$ at the earliest of:

1. At least once per 31 Effective Full Power Days, or

2. Upon reaching equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q{}^M(X,Y,Z)$ was last determined[2], or

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

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(1) No additional uncertainties are required in the following equations for $F_Q{}^M(X,Y,Z)$ because the limits include uncertainties.

(2) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

c. Performing the following calculations:

1. For each core location, calculate the % margin to the maximum allowable design as follows:

$$\% \text{ Operational Margin} = \left(1 - \frac{F_Q{}^M(X,Y,Z)}{[F_Q{}^L(X,Y,Z)]^{OP}}\right) \times 100\%$$

$$\% \text{ RPS Margin} = \left(1 - \frac{F_Q{}^M(X,Y,Z)}{[F_Q{}^L(X,Y,Z)]^{RPS}}\right) \times 100\%$$

where $[F_Q{}^L(X,Y,Z)]^{OP}$ and $[F_Q{}^L(X,Y,Z)]^{RPS}$ are the Operational and RPS design peaking limits defined in the COLR.

2. Find the minimum Operational Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then either of the following actions shall be taken:

(a) Within 15 minutes:

(1) Control the AFD to within new AFD limits that are determined by:

$$(\text{AFD Limit})_{\text{negative}}^{\text{reduced}} = (\text{AFD Limit})_{\text{negative}}^{\text{COLR (3)}} - \text{MARGIN}_{\text{OP}}^{\text{MIN}}$$

$$(\text{AFD Limit})_{\text{positive}}^{\text{reduced}} = (\text{AFD Limit})_{\text{positive}}^{\text{COLR (3)}} - \text{MARGIN}_{\text{OP}}^{\text{MIN}}$$

where $\text{MARGIN}_{\text{OP}}^{\text{MIN}}$ is the minimum margin from 4.2.2.2.c.1, and

(2) Within 8 hours, reset the AFD alarm setpoints to the modified limits of 4.2.2.2.c.2.a, or

(b) Comply with the ACTION requirements of Specification 3.2.2, treating the margin violation in 4.2.2.2.c.1 above as the amount by which $F_Q{}^{MA}$ is exceeding its limit.

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(3) Defined and specified in the COLR per Specification 6.9.1.9.

SURVEILLANCE REQUIREMENTS (Continued)

3. Find the minimum RPS Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then the following action shall be taken:

   Within 72 hours, reduce the K_1 value for OTΔT by:

   $$K_1 \text{ adjusted} = K_1^{(4)} - [\text{KSLOPE}^{(3)} \times \text{Margin}^{\min}_{\text{RPS}}] \text{ absolute value}$$

   where $\text{MARGIN}^{\min}_{\text{RPS}}$ is the minimum margin from 4.2.2.2.c.1.

d. Extrapolating[5] at least two measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

   $[F_Q{}^M(X,Y,Z)]$ (extrapolated) $\geq [F_Q{}^L(X,Y,Z)]^{OP}$ (extrapolated), and

   $$\frac{[F_Q{}^M(X,Y,Z)] \text{ (extrapolated)}}{[F_Q{}^L(X,Y,Z)]^{OP} \text{ (extrapolated)}} > \frac{[F_Q{}^M(X,Y,Z)]}{[F_Q{}^L(X,Y,Z)]^{OP}}$$

   or

   $[F_Q{}^M(X,Y,Z)]$ (extrapolated) $\geq [F_Q{}^L(X,Y,Z)]^{RPS}$ (extrapolated), and

   $$\frac{[F_Q{}^M(X,Y,Z)] \text{ (extrapolated)}}{[F_Q{}^L(X,Y,Z)]^{RPS} \text{ (extrapolated)}} > \frac{[F_Q{}^M(X,Y,Z)]}{[F_Q{}^L(X,Y,Z)]^{RPS}}$$

   either of the following actions shall be taken:

   1. $F_Q{}^M(X,Y,Z)$ shall be increased by 2 percent over that specified in 4.2.2.2.a, and the calculations of 4.2.2.2.c repeated, or

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(3) Defined and specified in the COLR per Specification 6.9.1.9.

(4) K_1 value from Table 2.2-1.

(5) Extrapolation of $F_Q{}^M$ for the initial flux map taken after reaching equilibrium conditions is not required since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F_Q{}^M$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

SURVEILLANCE REQUIREMENTS (Continued)

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.2.2.c.1 shall be performed no later than the time at which the margin in 4.2.2.2.c.1 is extrapolated to be equal to zero.

e. The limits in Specifications 4.2.2.2.c and 4.2.2.2.d are not applicable in the following core plane 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.

4.2.2.3 When a full core power distribution map is taken for reasons other than meeting the requirements of Specification 4.2.2.2, an overall $F_Q^M(X,Y,Z)$ shall be determined, then increased by 3% to account for manufacturing tolerances, further increased by 5% to account for measurement uncertainty, and further increased by the radial-local peaking factor to obtain a maximum local peak. This value shall be compared to the limit in Specification 3.2.2.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}(X,Y)$ shall be limited by imposing the following relationship:

$F_{\Delta H}{}^{M}(X,Y) \leq [F_{\Delta H}{}^{L}(X,Y)]^{LCO}$

where: $F_{\Delta H}{}^{M}(X,Y)$ - the measured radial peak.

$[F_{\Delta H}{}^{L}(X,Y)]^{LCO}$ - the maximum allowable radial peak as defined in Core Operating Limits Report (COLR).

APPLICABILITY: MODE 1.

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding its limit:

a. Within 2 hours, reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH%[1] for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds the limit, and

b. Within 6 hours either:

   1. Restore $F_{\Delta H}{}^{M}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH% for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds that limit, and

c. Within 72 hours of initially being outside the limit of Specification 3.2.3, either:

   1. Restore $F_{\Delta H}{}^{M}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

   2. Perform the following actions:

---

(1) RRH is the amount of THERMAL POWER reduction required to compensate for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

ACTION:

(a) Reduce the OTΔT K_1 term in Table 2.2-1 by at least TRH[2] for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds the limit, and

(b) Verify through incore mapping that $F_{\Delta H}{}^{M}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a., or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

---

(2) TRH is the amount of OTΔT K_1 setpoint reduction required to compensate for each 1% that $F_{\Delta H}{}^{M}(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

d. 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. and/or c.2 above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}{}^{M}(X,Y)$ is demonstrated, through incore flux mapping, to be within the Limit specified in the COLR prior to exceeding the following THERMAL POWER levels:

1. 50% of RATED THERMAL POWER,

2. 75% of RATED THERMAL POWER, and

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F_{\Delta H}{}^{M}(X,Y)$ shall be evaluated to determine whether $F_{\Delta H}(X,Y)$ 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. Measuring $F_{\Delta H}{}^{M}(X,Y)$ according to the following schedule:

1. Upon reaching equilibrium conditions after exceeding 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_{\Delta H}{}^{M}(X,Y)$ was last determined[3], or

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

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

c. Performing the following calculations:

1. For each location, calculate the % margin to the maximum allowable design as follows:

---

(3) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

$$\% \ F_{\Delta H} \text{ Margin} = \left(1 - \frac{F_{\Delta H}{}^{M}(X,Y)}{[F_{\Delta H}{}^{L}(X,Y)]^{surv}}\right) \times 100\%$$

No additional uncertainties are required for $F_{\Delta H}{}^{M}(X,Y)$, because $[F_{\Delta H}{}^{L}(X,Y)]^{surv}$ includes uncertainties.

2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above. If any margin is less than zero, comply with the ACTION requirements of Specification 3.2.3 as if $[F_{\Delta H}{}^{L}(X,Y)]^{surv}$ is the same as $F_{\Delta H}{}^{L}(X,Y)]^{LCO}$.

d. Extrapolating[4] at least two measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$F_{\Delta H}{}^{M}(X,Y)$ (extrapolated) $\geq [F_{\Delta H}{}^{L}(X,Y)^{surv}$ (extrapolated) and

$$\frac{F_{\Delta H}{}^{M}(X,Y) \quad \text{(extrapolated)}}{[F_{\Delta H}{}^{L}(X,Y)]^{surv}\text{(extrapolated)}} > \frac{F_{\Delta H}{}^{M}(X,Y)}{[F_{\Delta H}{}^{L}(X,Y)]^{surv}}$$

either of the following actions shall be taken:

1. $F_{\Delta H}{}^{M}(X,Y)$ shall be increased by 2 percent over that specified in 4.2.3.2.a, and the calculations of 4.2.3.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.3.2.c shall be performed no later than the time at which the margin in 4.2.3.2.c is extrapolated to be equal to zero.

---

(4) Extrapolation of $F_{\Delta H}{}^{M}$ for the initial flux map taken after reaching equilibrium conditions is not required since the initial flux map establishes the baseline measurement for future trending.

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 1.02 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 Exception 3.10.2.

**Not applicable until calibration of the excore detectors is completed subsequent to refueling.

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 1.02, 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 a 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},

b. Pressurizer Pressure, and

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1.

ACTION:

a. With either of the parameters identified in 3.2.5a. and b. above 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.

b. With the combination of Reactor Coolant System total flow rate and THERMAL POWER within the region of restricted operation specified on Figure 3.2.1, within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by Figure 3.2-1.

c. With the combination of RCS total flow rate and THERMAL POWER within the region of prohibited operation specified on Figure 3.2-1:

1. Within 2 hours either:

   a) Restore the combination of RCS total flow rate and THERMAL POWER to within the region of permissible operation,

   b) Restore the combination of Reactor Coolant System total flow rate and THERMAL POWER to within the region of restricted operation and comply with action b. above, or

   c) 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.

2. Within 24 hours of initially being within the region of prohibited operation specified in Figure 3.2-1, verify that the combination of THERMAL POWER and RCS total flow rate are restored to within the regions of permissible or restricted operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

3/4.2.5 DNB PARAMETERS

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be measured by averaging the indications (meter or computer) of the operable channels and verified to be within their limits at least once per 12 hours.

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

4.2.5.3 The RCS total flow rate shall be determined by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	INDICATION	# OPERABLE CHANNELS	LIMITS*
Indicated Reactor Coolant System T_{avg}	meter	4	≤ 590.5°F
	meter	3	≤ 590.2°F
	computer	4	≤ 591.0°F
	computer	3	≤ 590.8°F
Indicated Pressurizer Pressure**	meter	4	≥ 2226.5 psig
	meter	3	≥ 2229.8 psig
	computer	4	≥ 2221.7 psig
	computer	3	≥ 2224.2 psig
Reactor Coolant System Total Flow Rate			Figure 3.2-1

*Limits applicable during four-loop operation.

**Limits 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% RATED THERMAL POWER.

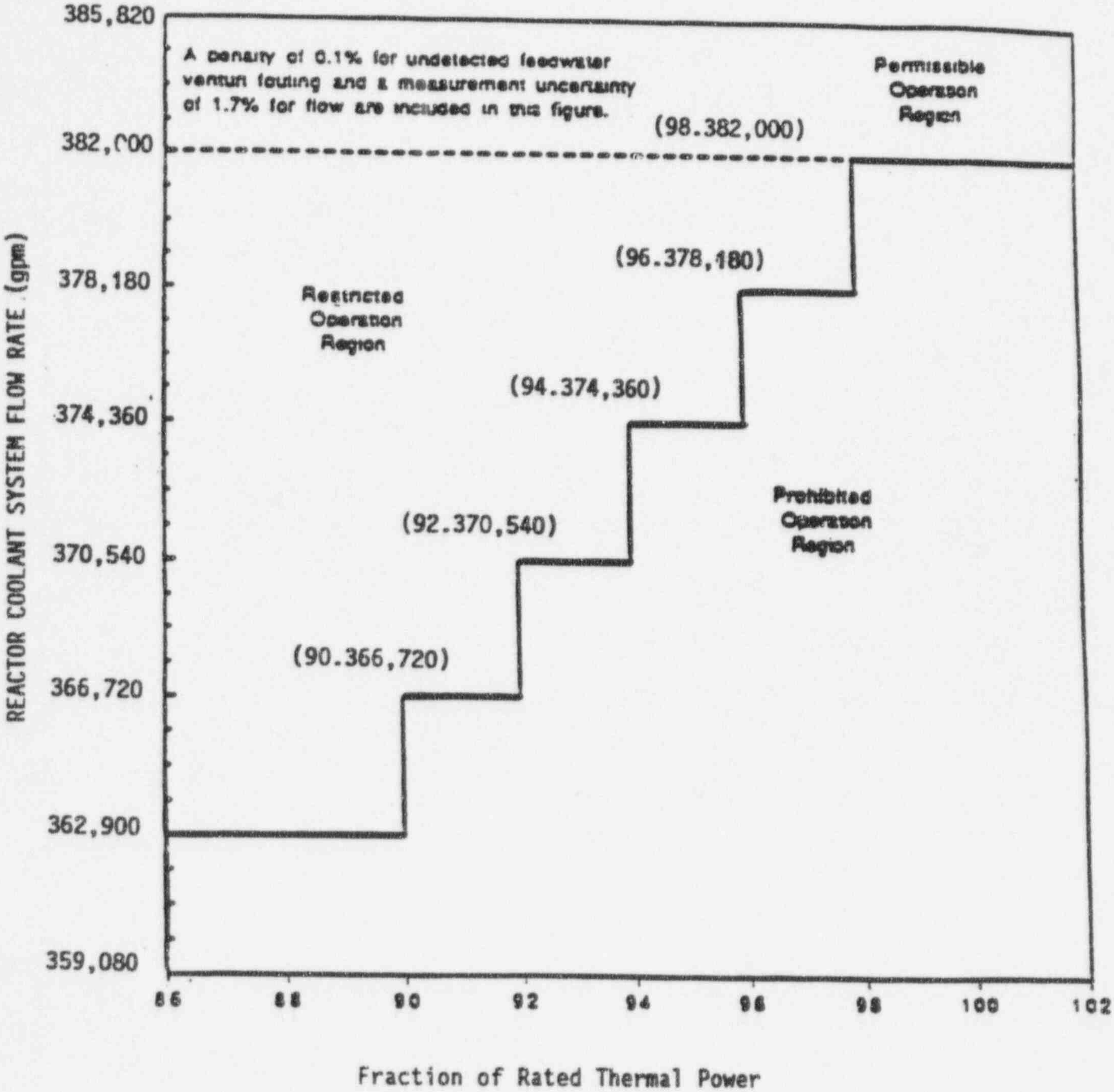

FIGURE 3.2 - 1. REACTOR COOLANT SYSTEM TOTAL FLOW RATE VERSUS RATED THERMAL POWER - FOUR LOOPS IN OPERATION

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System Instrumentation channel and interlock 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.

4.3.1.3 The response time of RTDs associated with the Reactor Trip System shall be demonstrated to be within their limits (see note 2 to Table 3.3-2) at least once per 18 months.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux - High Setpoint	4	2	3	1, 2	2
Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
5. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
c. Shutdown	2	0	1	3, 4, and 5	5
6. Overtemperature ΔT					
Four Loop Operation	4	2	3	1, 2	6
Three Loop Operation	(**)	(**)	(**)	(**)	(**)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	6
Three Loop Operation	(**)	(**)	(**)	(**)	(**)
8. Pressurizer Pressure-Low	4	2	3	1	6
9. Pressurizer Pressure--High	4	2	3	1, 2	6
10. Pressurizer Water Level--High	3	2	2	1	6
11. Low Reactor Coolant Flow					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6
12. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6

TABLE 3.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Undervoltage-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6
14. Underfrequency-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6
15. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1	6
b. Turbine Stop Valve Closure	4	4	1	1	11
16. Safety Injection Input from ESF	2	1	2	1, 2	7
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
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Low Setpoint Power Range Neutron Flux, P-10	4	2	3	1, 2	8
e. Turbine Impulse Chamber Pressure, P-13	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
18. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10
19. Automatic Trip and Interlock Logic	2	1	2	1, 2	7
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)
TABLE NOTATION

*With the Reactor Trip System breakers in the closed position, the Control Rod Drive System capable of rod withdrawal.

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

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

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

ACTION STATEMENTS

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, 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, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

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

a. The inoperable channel is placed in the tripped condition within 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 and Specification 4.3.2.1.

ACTION 7 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

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 the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

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

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 12 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 9. 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	≤ 0.5 second (1)
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Intermediate Range, Neutron Flux	N.A.
5. Source Range, Neutron Flux	N.A.
6. Overtemperature ΔT	≤ 10.0 seconds (1)(2)
7. Overpower ΔT	≤ 10.0 seconds (1)(2)
8. Pressurizer Pressure--Low	≤ 2.0 seconds
9. Pressurizer Pressure--High	≤ 2.0 seconds
10. Pressurizer Water Level--High	N.A.

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

(2) The ≤ 10.0 second response time includes a 6.5 second delay for the RTDs mounted in thermowells.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	$\leq$ 1.0 second
b. Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 second
12. Steam Generator Water Level--Low-Low	$\leq$ 3.5 seconds
13. Undervoltage-Reactor Coolant Pumps	< 1.5 seconds
14. Underfrequency-Reactor Coolant Pumps	< 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.

TABLE 4.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R (11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1###, 2
5. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),Q(9)	N.A.	N.A.	2##, 3, 4, 5
6. Overtemperature ΔT	S	R(15)	Q	N.A.	N.A.	1, 2
7. Overpower ΔT	S	R(15)	Q	N.A.	N.A.	1, 2
8. Pressurizer Pressure--Low	S	R	Q	N.A.	N.A.	1
9. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2
10. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1
11. Low Reactor Coolant Flow	S	R	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level--Low-Low	S	R	Q	N.A.	N.A.	1, 2
13. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
14. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q	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)	N.A	N.A.	N.A.	2##
b. Power Range Neutron Flux, P-8	N.A.	R(4)	N.A	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
c. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	N.A	N.A.	N.A.	1, 2
d. Turbine Impulse Chamber Pressure, P-13	N.A.	R	N.A	N.A.	N.A.	1
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 12)	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. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	M(13),R(14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With the Reactor Trip System breakers closed and the Control Rod Drive System capable of rod withdrawal.

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

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

(1) - If not performed in previous 31 days.

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

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

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

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

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

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

(8) - Deleted.

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

(10) - Setpoint verification is not required.

TABLE 4.3-1 (Continued)

TABLE NOTATION

(11) - The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function.

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

(13) - Prior to placing breaker in service, a local manual shunt trip shall be performed.

(14) - The automatic undervoltage trip capability shall be verified operable.

(15) - Overtemperature setpoint, overpower setpoint, and T_{avg} channels require an 18 month channel calibration. Calibration of the ΔT channels is required at the beginning of each cycle upon completion of the precision heat balance. RCS loop ΔT values shall be determined by precision heat balance measurements at the beginning of each cycle.

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

b. With an ESFAS Instrumentation channel 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 the 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, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3	15
d. Pressurizer Pressure - Low-Low	4	2	3	1, 2, 3#	19
e. Steam Line Pressure-Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (continued)					
b. Phase "B" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	4	2	3	1, 2, 3	16
c. Purge and Exhaust Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) System	2	1	2	1, 2, 3	22
2) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
d. Negative Steam Line Pressure Rate - High					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	3##	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)
e. Steam Line Pressure - Low					
Four Loops Operating	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	15
Three Loops Operating	(**)	(**)	(**)	(**)	(**)

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 & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relay	2	1	2	1, 2	21
b. Steam Generator Water Level--High-High	3/stm. gen.	2/stm. gen. in any operating stm gen.	2/stm. gen. in each operating stm. gen.	1, 2	15
c. Doghouse Water Level (Feedwater Isolation Only)	3/train/ Doghouse	2/train/ Doghouse	2/train/ Doghouse	1, 2	25
6. Containment Pressure Control System	8(4/train)	4/train	8	1, 2, 3, 4	26

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
7. Auxiliary Feedwater					
a. Manual Initiation	2	1	2	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Stm. Gen. Water Level-Low-Low					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen in each operating stm. gen	1, 2, 3	19
d. Auxiliary Feedwater Suction Pressure - Low	2/motor driven pump	2/pump	2 of the same train/pump	1, 2, 3	24
(Suction Supply Automatic Realignment)	4/turbine driven pump	2/pump	2 of the same train/pump	1, 2, 3	24
e. Safety Injection Start Motor-Driven Pumps					

See Item 1. above for all Safety Injection initiating functions and requirements

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Auxiliary Feedwater (continued)					
f. Station Blackout (Note 1) Start Motor-Driven Pumps and Turbine-Driven Pump					
1) 4 kV Loss of Voltage	3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19
2) 4 kV Degraded Voltage	3/Bus	2/Bus Either Bus	2/Bus	1, 2, 3	19
g. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	2-1/MFWP	2-1/MFWP	2-1/MFWP	1, 2#	27
8. Automatic Switchover to Recirculation RWST Level	3	2	2	1, 2, 3	15b
9. Loss of Power					
a. 4 kV Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15a
b. 4 kV Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	15a
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22
d. Steam Generator Level, P-14	3/stm gen.	2/stm gen. in any operating stm gen.	2/stm gen. in each operating stm gen.	1, 2, 3	20

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

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

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

Note 1: Turbine driven auxiliary feedwater pump will not start on a blackout signal coincident with a safety injection signal.

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 12 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 15a With the number of OPERABLE channels less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours. With more than one channel inoperable, enter Specification 3.8.1.1.

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

ACTION 16 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

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

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

b. The Minimum Channels OPERABLE requirement i[illegible], 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 Specification 4.3.2.1.

ACTION 20 - 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 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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

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

ACTION 24 - With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.7.1.2. With the channels associated with more than one auxiliary feedwater pump inoperable, immediately declare the associated auxiliary feedwater pumps inoperable and take the action required by Specification 3.7.1.2.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 25 - With one of the two trains of doghouse water level instrumentation inoperable (less than the minimum required number of channels operable), restore the inoperable train to operable status in 72 hours. After 72 hours with one train inoperable, or within one hour with 2 trains inoperable, monitor doghouse water level in the affected doghouse continuously until both trains are restored to operable status.

ACTION 26 - With any of the eight channels inoperable, place the inoperable channel(s) in the start permissive mode within one hour and apply the applicable action statement (Containment Spray - T.S. 3.6.2, Containment Air Return/Hydrogen Skimmer - T.S. 3.6.5.6).

ACTION 27 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Safety Injection, Reactor Trip, Feedwater Isolation, Component Cooling Water, Start Diesel Generators, and Nuclear Service Water.		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High	≤ 1.1 psig	≤ 1.2 psig
d. Pressurizer Pressure--Low-Low	≥ 1845 psig	≥ 1835 psig
e. Steam Line Pressure - Low	≥ 775 psig	≥ 755 psig
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	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.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High	$\leq$ 2.9 psig	$\leq$ 3.0 psig
c. Purge and Exhaust Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 2.9 psig	≤ 3.0 psig
d. Negative Steam Line Pressure Rate - High	≤ 100 psi with a rate/lag function time constant ≥ 50 seconds	≤ 120 psi with a rate/lag function time constant ≥ 50 seconds
e. Steam Line Pressure - Low	≥ 775 psig	≥ 755 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level--High-High (P-14)	≤ 82% of narrow range instrument span each steam generator	≤ 83% of narrow range instrument span each steam generator
c. Doghouse Water Level-High (Feedwater Isolation Only)	12"	13"
6. Containment Pressure Control System		
Start Permissive/Termination (SP/T)	0.3 ≤ SP/T ≤ 0.4 PSIG	0.25 ≤ SP/T ≤ 0.45 PSIG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	≥ 12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40.0% of span at 100% of RATED THERMAL POWER.	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 39.0% of span at 100% of RATED THERMAL POWER.
2) Start Turbine-Driven Pumps	12% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 40.0% of span at 100% of RATED THERMAL POWER.	≥ 11% of span from 0 to 30% of RATED THERMAL POWER, increasing linearly to ≥ 39.0% of span at 100% of RATED THERMAL POWER.
d. Auxiliary Feedwater Suction Pressure - Low (Suction Supply Automatic Realignment)	≥ 2 psig	≥ 1 psig
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TABLE 3.3-4 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Auxiliary Feedwater (continued)		
f. Station Blackout - Start Motor-Driven Pumps and Turbine-Driven Pump (Note 1)		
1) 4 kV Loss of Voltage	3174 ± 45 volts with a 8.5 ± 0.5 second time delay	≥ 3122 volts
2) 4 kV Degraded Voltage	≥ 3678.5 volts with ≤ 11 second with SI and ≤ 600 second without SI time delays	≥ 3661 volts
g. Trip of Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.
8. Automatic Switchover to Recirculation RWST Level	≥ 90 inches	≥ 80 inches
9. Loss of Power		
a. 4 kV Loss of Voltage	3174 ± 45 volts with a 8.5 ± 0.5 second time delay	≥ 3122 volts
b. 4 kV Degraded Voltage	≥ 3678.5 volts with ≤ 11 second with SI and ≤ 600 second without SI time delays	≥ 3661 volts
10. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	≤ 1955 psig	≤ 1965 psig
b. T_{avg}, P-12	≥ 553°F	≥ 551°F
c. Reactor Trip, P-4	N.A.	N.A.
d. Steam Generator Level, P-14	See Item 5b. above for all Trip Setpoints and Allowable Values.	

Note 1: The turbine driven pump will not start on a blackout signal coincident with a safety injection signal.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Containment Isolation	
Phase "A" Isolation	N.A.
Phase "B" Isolation	N.A.
Purge and Exhaust Isolation	N.A.
d. Steam Line Isolation	N.A.
e. Feedwater Isolation	N.A.
f. Auxiliary Feedwater	N.A.
g. Nuclear Service Water	N.A.
h. Component Cooling Water	N.A.
i. Reactor Trip (from SI)	N.A.
j. Start Diesel Generators	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27^{(1)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
i. Start Diesel Generators	≤ 11

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low-Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water System	$\leq 76^{(1)}/65^{(3)}$
h. Component Cooling Water	$\leq 76^{(1)}/65^{(3)}$
i. Start Diesel Generators	≤ 11
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12^{(3)}/22^{(4)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 12
d. Containment Isolation-Phase "A"(2)	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	≤ 4
f. Auxiliary Feedwater(5)	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Steam Line Isolation	≤ 10
i. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
j. Start Diesel Generators	≤ 11

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
5. Containment Pressure-High-High	
a. Containment Spray	≤ 45
b. Containment Isolation-Phase "B"	N.A.
c. Steam Line Isolation	≤ 10
6. Steam Generator Water Level-High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ 12
7. Steam Generator Water Level Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-driven Auxiliary Feedwater Pump	≤ 60
8. Negative Steam Line Pressure Rate - High	
Steam Line Isolation	≤ 10
9. Start Permissive	
Containment Pressure Control System	N.A.
10. Termination	
Containment Pressure Control System	N.A.
11. Auxiliary Feedwater Suction Pressure - Low	
Auxiliary Feedwater Pumps (Suction Supply Automatic Realignment)	≤ 13
12. RWST Level	
Automatic Switchover to Recirculation	≤ 60

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
13. Station Blackout	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine-Drive Auxiliary Feedwater Pump [(6)]	≤ 60
14. Trip of Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
15. Loss of Power	
a. 4 kV Loss of Voltage	≤ 11
b. 4 kV Degraded Voltage	≤ 11 with SI, and ≤ 600 without SI

TABLE 3.3-5 (Continued)

TABLE NOTATION

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

(2) Valves 1KC305B and 1KC315B are exceptions to the response times listed in the table. The following response times in seconds are the required values for these valves for the initiating signal and function indicated:

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

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

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

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

(6) The turbine driven pump does not start on a blackout signal coincident with a safety injection signal.

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, Component Cooling Water, Start Diesel Generators, and Nuclear 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	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure--Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
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-High	S	R	Q	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 and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
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.						

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								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Negative Steam Line Pressure Rate-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	3
e. Steam Line Pressure--Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	Q	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Doghouse Water Level-High (Feedwater Isolation Only)	S	N.A	N.A	R	N.A.	N.A.	N.A.	1, 2
6. Containment Pressure Control System Start Permissive/ Termination	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	Q	N.A.	N.A.	N.A	N.A.	1, 2, 3
d. Auxiliary Feedwater Suction Pressure-Low	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
f. Station Blackout	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
8. Automatic Switchover to Recirculation								
RWST Level	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
9. Loss of Power								
a. 4 kV Loss of Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3, 4
b. 4 kV Degraded Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low, Low T_{avg}, P-12	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Level, P-14	See Item 5b for all surveillance requirements							

TABLE NOTATION

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

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 exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the action shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

MONITOR	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment Atmosphere Gaseous Radioactivity-High (Low Range-EMF-39)	1	1	1, 2, 3, 4	***	26
2. Spent Fuel Pool Radioactivity-High (EMF-42)	1	1	**	$\leq 1.7 \times 10^{-4}$ μCi/ml	30
3. Criticality-Radiation Level (1EMF-17)	1	1	*	$\leq$ 15 mR/hr	28
4. Gaseous Radioactivity-RCS Leakage Detection (Low Range - EMF-39)	N.A.	1	1, 2, 3, 4	N.A.	29
5. Particulate Radioactivity-RCS Leakage Detection (Low Range - EMF-38)	N.A.	1	1, 2, 3, 4	N.A.	29

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

MONITOR	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
6. Control Room Air Intake Radioactivity-High (EMF-43a and 43b)	1 per station	2 per station	All	$\leq 3.4 \times 10^{-4}$ $\mu Ci/ml$	27

TABLE NOTATION

* - With fuel in the fuel storage areas or fuel building.

** - With irradiated fuel in the fuel storage areas or fuel building.

*** - Must satisfy the requirements of McGuire Selected Licensee Commitment 16.11-6.

ACTION STATEMENTS

ACTION 26 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge 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 Ventilation System outside air intake which contains the inoperable instrumentation.

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 pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel building.

ACTION 29 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 30 - With less than the minimum channels OPERABLE requirement, operation may continue provided the Fuel Handling Ventilation Exhaust System requirements of Specification 3/4.9.11 are met.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

MONITOR	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES REQUIRING SURVEILLANCE
1. Containment Atmosphere Gaseous Radioactivity-High (Low Range-EMF-39)	S	R	Q	1, 2, 3, 4
2. Spent Fuel Pool Ventilation Radioactivity-High (EMF-42)	S	R	Q	**
3. Criticality-High Radiation Level (1EMF-17)	S	R	Q	*
4. Gaseous Radioactivity-RCS Leakage Detection (Low Range-EMF-39)	S	R	Q	1, 2, 3, 4
5. Particulate Radioactivity-RCS Leakage Detection (Low Range-EMF-38)	S	R	Q	1, 2, 3, 4
6. Control Room Air Intake Radioactivity-High (EMF-43a and EMF-43b)	S	R	Q	All

TABLE NOTATION

* - With fuel in the fuel handling area.

** - With irradiated fuel in the fuel handling 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,

b. Monitoring the QUADRANT POWER TILT RATIO, or

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

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

b. Monitoring the QUADRANT POWER TILT RATIO, or

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

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9
REMOTE SHUTDOWN MONITORING INSTRUMENTATION

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

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required 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 SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status according to b.1 and b.2 below:

   b.1 Instruments 1-15: within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

   b.2 Instruments 16 and 17: according to Technical Specification 3.7.4.a.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Temperature - T_{HOT} and T_{COLD} (Wide Range)	$2/T_{HOT}$ $2/T_{COLD}$	$1/T_{HOT}$ $1/T_{COLD}$
3. Reactor Coolant Pressure - Wide Range	2	1
4. Pressurizer Water Level	2	1
5. Steam Line Pressure	2/steam generator	1/steam generator
6. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
7. Refueling Water Storage Tank Water Level	2	1
8. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator
9. Reactor Coolant System Subcooling Margin Monitor	2	1
10. Containment Water Level (Wide Range)	2	1
11. In Core Thermocouples	4/core quadrant	2/core quadrant
12. Containment Atmosphere - High Range Monitor (EMF-51a or 51b)	1	1
13. Reactor Vessel Level Instrumentation		
a. Dynamic Head (D/P) Range	2	1
b. Lower Range	2	1
14. Neutron Flux - Wide Range	2	1
15. Containment Hydrogen Concentration	2	1
16. Diesel Generator Cooling Water Heat Exchanger RN Flow*	1/diesel generator	1/diesel generator
17. Containment Spray Heat Exchanger RN Flow*	1/train	1/train

*Not applicable if the associated outlet valve is set to its flow balance position with power removed or if the associated outlet valve's flow balance position is fully open.

TABLE 4.3-7
ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Temperature - T_{HOT} and T_{COLD} (Wide Range)	M	R
3. Reactor Coolant Pressure - Wide Range	M	R
4. Pressurizer Water Level	M	R
5. Steam Line Pressure	M	R
6. Steam Generator Water Level - Narrow Range	M	R
7. Refueling Water Storage Tank Water Level	M	R
8. Auxiliary Feedwater Flow Rate	M	R
9. Reactor Coolant System Subcooling Margin Monitor	M	R
10. Containment Water Level (Wide Range)	M	R
11. In Core Thermocouples	M	R
12. Containment Atmosphere - High Range Monitor (EMF-51a or 51b)	M	R
13. Reactor Vessel Level Instrumentation		
a. Dynamic Head (D/P) Range	M	R
b. Lower Range	M	R
14. Neutron Flux - Wide Range	M	R
15. Containment Hydrogen Concentration	M	R
16. Diesel Generator Cooling Water Heat Exchanger RN Flow	M	R
17. Containment Spray Heat Exchanger RN Flow	M	R

3/4.3.3.7 DELETED

3/4.3.3.8 DELETED

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 3.3-13.

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-13

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Not Used			
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitor	1 per station	**	1
b. Oxygen Monitors	2 per station	**	2
3. Not Used			
4. Not Used			

**During WASTE GAS HOLDUP SYSTEM operation.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

ACTION 2 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With two channels inoperable, be in at least HOT STANDBY within 6 hours.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

MONITOR	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES REQUIRING SURVEILLANCE
1. Not Used				
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System				
a. Hydrogen Monitor	D	Q(1)	M	**
b. Oxygen Monitor	D	Q(2)	M	**
c. Oxygen Monitor (alternate)	D	Q(2)	M	**
3. Not Used				
4. Not Used				

TABLE NOTATION

**During WASTE GAS HOLDUP SYSTEM operation.

(1) The CHANNEL CALIBRATION shall include the use of standard gas samples corresponding to alarm setpoints in accordance with the manufacturer's recommendations.

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

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

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

SURVEILLANCE REQUIREMENT

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

---

*See Special Test Exception 3.10.4.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

APPLICABILITY: MODE 3

ACTION:

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

b. With less than the above required reactor coolant loops in operation, restore the required loops to operation within 72 hours or open the Reactor Trip System breakers.

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

SURVEILLANCE REQUIREMENT

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

4.4.1.2.2 At least the above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

e. RHR Loop A, and

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

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

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

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*A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless: (1) the pressurizer water volume is less than 92% (1600) cubic feet, or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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

SURVEILLANCE REQUIREMENT

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

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

4.4.1.3.3 At least once 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 12%.

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

ACTION:

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

b. With no RHR 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 REQUIREMENT

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.

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

#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 300°F unless: (1) the pressurizer water level is less than 92% (1600 cubic feet), or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperature.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required 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 REQUIREMENT

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

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig + 3%, - 2%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

*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 level of less than or equal to 92% (1600 cubic feet), and at least two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

c. With two PORVs inoperable due to causes other than excessive leakage, within 1 hour either restore the PORVs to OPERABLE status or close the associated block valves and remove power from the block valves; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If the block valves have been closed and power has been removed, restore at least one PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With three PORVs inoperable due to causes other than excessive leakage, within 1 hour either restore at least one PORV to OPERABLE status or close the associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV switch in the "close" position and remove power from its associated solenoid valve (do no enter action statement b for the resulting inoperable PORV); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

LIMITING CONDITION FOR OPERATION

f. With two block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place their associated PORV switches in the "close" position (do not enter action statement c for the resulting inoperable PORVs); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If the PORV switches have been placed in the "close" position, restore at least one block valve to OPERABLE status within 72 hours; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

g. With three block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place their associated PORV switches in the "close" position (do not enter action statement d for the resulting inoperable PORVs). Restore at least one block valve to OPERABLE status within the next hour; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel during MODE 3 or MODE 4 when the temperature of all RCS cold legs is greater than 300F with the block valve closed.

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 in order to meet the requirements of ACTION a., b., c., or d. in Specification 3.4.4.

4.4.4.3. The emergency power supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive power from the normal (air) supply to the emergency (nitrogen) supply.

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

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

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum 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 4.4.5.4.a.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. In addition to the 3% sample, all F* tubes will be inspected.

d. 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 following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

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

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

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

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

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

    4) A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

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 or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve 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 or sleeve;

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

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

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

6) Repair Limit means the imperfection depth at or beyond which the tube or sleeve shall be removed from service by plugging or repaired by sleeving and is equal to 40% of the nominal tube or sleeve wall thickness. This definition does not apply to the area of the tubesheet region below the F* distance provided the tube is not degraded (i.e., no indications of cracking) within the F* distance. If a tube is sleeved due to degradation in the F* distance, then any defects in the tube below the sleeve will remain in service without repair.

The Babcock & Wilcox process (or method) equivalent to the method described in Topical Report BAW-2045(P)-A will be used.

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

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

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 after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) F* Distance is the distance into the tubesheet from the top face of the tubesheet or the top of the last hardroll, whichever is lower (further into the tubesheet) that has been conservatively chosen to be 2 inches.

11) F* TUBE is a tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of cracking) in the F* distance.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the repair 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 or repaired.

c. The results of inspections of F* tubes shall be reported to the Commission in a report, prior to the restart of the unit following the inspection. This report shall include:

1) Identification of F* tubes, and

2) Location and size of the degradation.

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 NOTATION:

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 deflective 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.	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.	N/A	N/A

S = 3 (N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generator 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. Either the Containment Floor and Equipment Sump Level System or the Flow Monitoring System, and

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

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

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

c. Containment Ventilation Condensate Drain Tank Level 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 primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

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

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

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

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

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 sump inventory or discharge at least once per 12 hours;

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

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

e. Monitoring the Reactor Head Flange Leakoff 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, and

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
NI60	Accumulator Discharge
NI71	Accumulator Discharge
NI59	Accumulator Discharge
NI70	Accumulator Discharge
NI82	Accumulator Discharge
NI94	Accumulator Discharge
NI81	Accumulator Discharge
NI93	Accumulator Discharge
NI134	Safety Injection (Hot Leg)
NI159	Safety Injection (Hot Leg)
NI156	Safety Injection (Hot Leg)
NI128	Safety Injection (Hot Leg)
NI124	Safety Injection (Hot Leg)
NI160	Safety Injection (Hot Leg)
NI157	Safety Injection (Hot Leg)
NI126	Safety Injection (Hot Leg)
NI129	Safety Injection (Hot Leg)
NI125	Safety Injection (Hot Leg)
NI165	Safety Injection/Residual Heat Removal (Cold Leg)
NI167	Safety Injection/Residual Heat Removal (Cold Leg)
NI169	Safety Injection/Residual Heat Removal (Cold Leg)
NI171	Safety Injection/Residual Heat Removal (Cold Leg)
NI175	Safety Injection/Residual Heat Removal (Cold Leg)
NI176	Safety Injection/Residual Heat Removal (Cold Leg)
NI180	Safety Injection/Residual Heat Removal (Cold Leg)
NI181	Safety Injection/Residual Heat Removal (Cold Leg)
ND1B*	Residual Heat Removal
ND2A*	Residual Heat Removal

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

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

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

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

At All Other Times:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

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

---

*Limit not applicable with T_{avg} less than or equal to 250°F.

**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 1.0 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross specific activity.

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 1.0 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours;

b. With the specific activity of the 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; and

c. The provisions of Specification 3.0.4 are not applicable.

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

With the specific activity of the reactor coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram of gross specific activity, 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.

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.

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

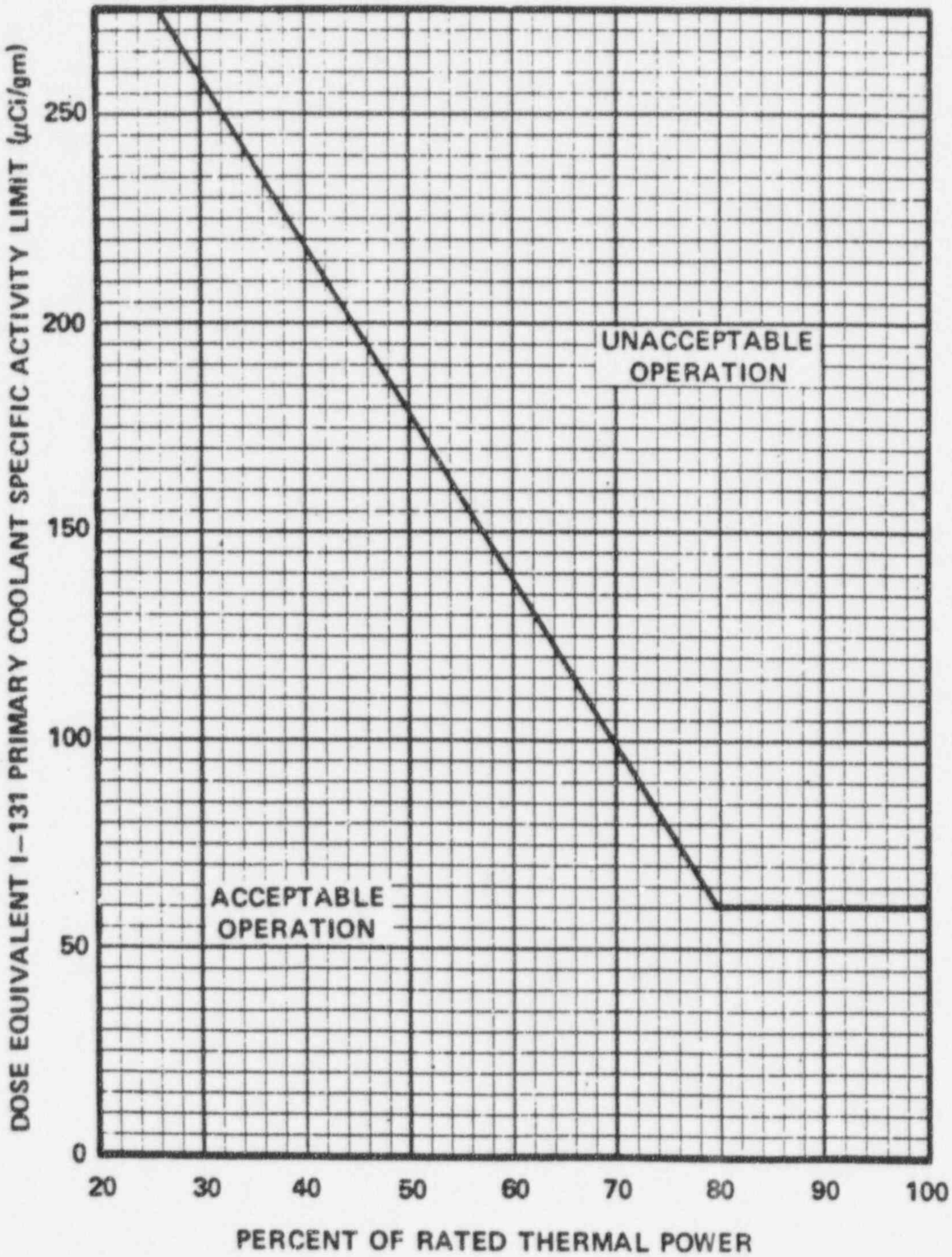


FIGURE 3.4-1
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

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 Specific Activity Determination**	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination***	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram 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 NOTATION

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

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

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the beta-gamma activity in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Maximum heatup rates as specified in Figure 3.4-2

b. Maximum cooldown rates as specified in Figure 3.4-3

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENT

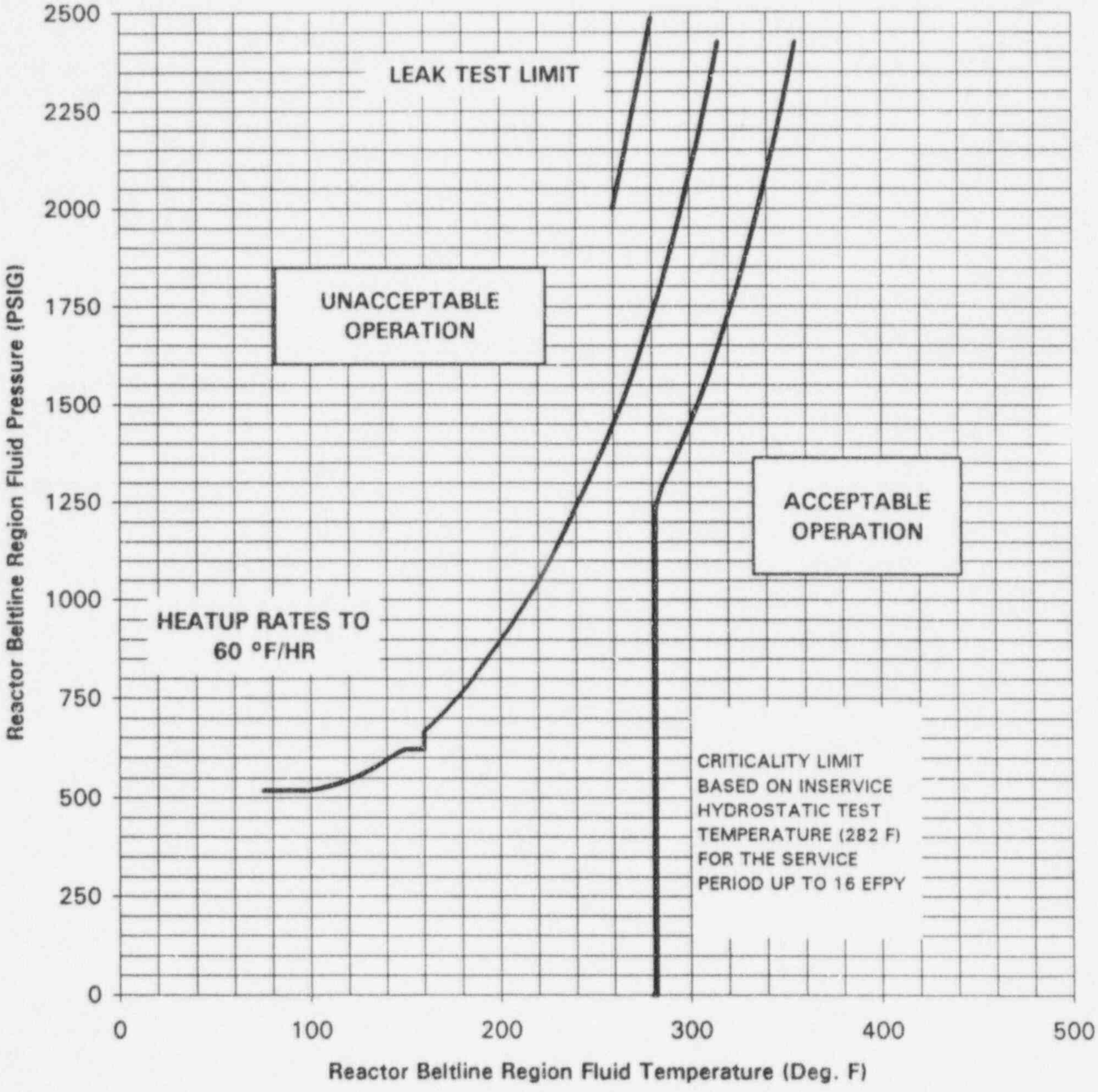
4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

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 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

LIMITING MATERIALS: LOWER SHELL LONGITUDINAL WELDS 3-442A and LOWER SHELL PLATE B5013-2

LIMITING ART AT 16 EFPY:

1/4-t, 149.5 deg. F
3/4-t, 102.0 deg. F



Reactor Coolant System Heatup Limitations
(Without margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the first 16 EFPY

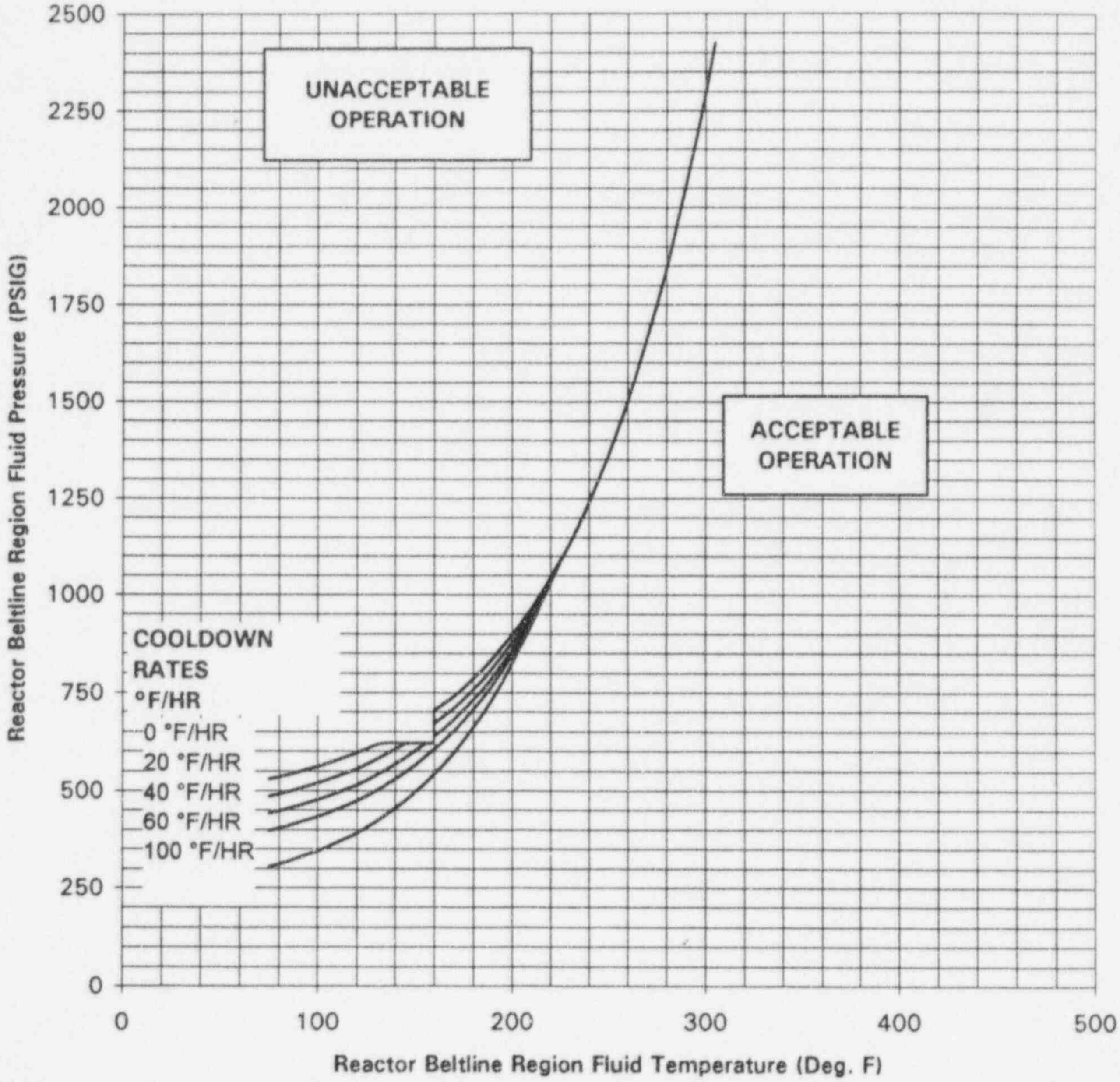
Figure 3.4-2

MATERIAL PROPERTY BASIS:

LIMITING MATERIALS:

LOWER SHELL LONGITUDINAL WELDS 3-442A and
LOWER SHELL PLATE B5013-2

LIMITING ART AT 16 EFPY:
1/4-t, 149.5 deg. F
3/4-t, 102.0 deg. F



RCS Cooldown Limitations,
Cooldown Rates up to 100 deg. F/HR
(Without Margins for Instrumenation Errors)
NRC REG GUIDE 1.99, REV. 2
Applicable for the First 16 EFPY

Figure 3.4-3

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 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITION FOR OPERATION

3.4.9.3 As a minimum, a Low Temperature Overpressure Protection (LTOP) System shall be OPERABLE as follows:

a. A maximum of one Centrifugal Charging (NV) pump or one Safety Injection (NI) pump capable of injecting into the Reactor Coolant System (RCS) with all remaining NV and NI pump motor circuit breakers open or the discharge of the remaining NV and NI pumps isolated from the RCS by at least 2 valves with power removed#

AND

b. All accumulators isolated

AND

c. One of the following conditions met:

1. Two PORVs with a lift setting of $\leq$ 385 psig

OR

2. The RCS depressurized with a vent of $\geq$ 2.75 square inches.

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

ACTION:

a. With two or more Charging (NV) or Safety Injection(NI) pumps capable of injecting into the RCS*, immediately initiate action to restore a maximum of one NI or one NV pump capable of injecting into the RCS.

Two Charging pumps (NV or NI) maybe capable of injecting into the RCS during pump swap operation for $\leq$ 15 Minutes.

* One Safety Injection pump and one Charging pump, or two Charging pumps may be operated concurrently provided:

1. RHR suction relief valve (ND-3) is OPERABLE, and the RHR suction isolation valves (ND-1 and ND-2) are open and one of the following conditions is met:

    a. RCS cold leg temperature is greater than 167° F or

    b. RCS cold leg temperature is greater than 107° F and cooldown rate is less than 20° F per hour.

OR

2. Two PORVs secured in the open position with their associated block valves open and power removed.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (continued)

b. With an accumulator not isolated, isolate the affected accumulator within 1 hour. If required action is not met, either:

   1. Depressurize the accumulator to less than the maximum RCS pressure for the existing cold leg per Specification 3/4.4.9 within 12 hours,

   OR

   2. Increase RCS cold leg temperature to greater than or equal to 300° F within 12 hours.

c. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days. If required action is not met, depressurize the RCS and vent through at least a 2.75 square inch vent within 8 hours.

d. With one PORV inoperable in MODES 5 or 6, suspend all operations which could lead to a water-solid pressurizer. Restore the inoperable PORV to OPERABLE status within 24 hours. If required action is not met, either:

   1. Ensure RCS temperature is greater than 167° F, and ND-3 is OPERABLE, and ND-1 and ND-2 are open within one hour.

   OR

   2. Depressurize the RCS and vent through at least a 2.75 square inch vent within 8 hours.

e. With the LTOP system inoperable for any reason other than a., b., c., or d. above, depressurize the RCS and vent through at least a 2.75 square inch vent within 8 hours.

f. In the event that either the PORVs or the RCS vent 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 circumstance initiating the transient, the effect of the PORVs or vent on the transient, and any corrective action necessary to prevent recurrence.

g. 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, at least once per 31 days;

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 Once every 12 hours*, verify that an RCS vent of ≥ 2.75 square inches is open when the vent is used for overpressure protection.

4.4.9.3.3 Once every 12 hours, verify that each accumulator is isolated and that only one NV or NI pump is capable of injecting into the RCS.

4.4.9.3.4 Once every 12 hours, verify that RHR suction isolation valves ND-1 and ND-2 are open when RHR suction relief valve ND-3 is being used for overpressure protection.

4.4.9.3.5 Once every 72 hours, verify that the PORV block valve is open for each required PORV.

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* 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.

A PORV secured in the open position may be used to meet this vent requirement provided that its associated block valve is open and power is removed.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: All MODES.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.4.11 REACTOR VESSEL HEAD VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.11 Two reactor vessel head vent paths, each consisting of two valves in series powered from emergency buses shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With one of the above reactor vessel head paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both of the above reactor vessel head vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least two of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

2. Verifying flow through the reactor vessel head vent paths during venting during COLD SHUTDOWN or REFUELING.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6870 and 7342 gallons,

c. A boron concentration between the LCO limits presented in the Core Operating Limits Report,

d. A nitrogen cover-pressure of between 585 and 639 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than the lower LCO limit presented in the Core Operating Limits Report, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System 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 Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

c. With one accumulator inoperable due to boron concentration less than the lower LCO limit presented in the Core Operating Limits Report and:

   1) The volume weighted average boron concentration of the accumulators equal to the lower LCO limit presented in the Core Operating Limits Report or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

   2) The volume weighted average boron concentration of the accumulators less than the lower LCO limit presented in the Core Operating Limits Report but greater than the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report, restore the inoperable

---

*Reactor Coolant System pressure above 1000 psig.

accumulator to OPERABLE status or return the volume weighted average boron concentration of the accumulators to greater than the lower LCO limit presented in the Core Operating Limits Report and enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

3) The volume weighted average boron concentration of the accumulators equal to the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report or less, return the volume weighted average boron concentration of the accumulators to greater than the minimum required to ensure post-LOCA subcriticality presented in the Core Operating Limits Report and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2) Verifying that each cold leg injection accumulator isolation valve is open.

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 not resulting from normal makeup by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that power to the isolation valve operator is disconnected; and

d. At least once per 18 months by verifying proper operation of the power disconnect circuit.

4.5.1.1.2 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
NI162A	Cold Leg Recirc.	Open*
NI121A	Hot Leg Recirc.	Closed
NI152B	Hot Leg Recirc.	Closed
NI183B	Hot Leg Recirc.	Closed
NI173A	RHR Pump Discharge	Open*
NI178B	RHR Pump Discharge	Open*
NI100B	SI Pump RWST Suction	Open
FW27A	RHR/RWST Suction	Open*
NI147A	SI Pump Mini flow	Open

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, unless the pumps and associated piping are in service or have been in service within 31 days, 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 interlock action of the RHR System from the Reactor Coolant System by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened.

*Valves may be realigned to place RHR System in service and for testing pursuant to Specification 4.4.6.2.2.

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

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

1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection actuation and automatic switchover to Containment Sump Recirculation test signals, 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.

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

1)	Centrifugal charging pump	$\geq$ 2347 psid,
2)	Safety Injection pump	$\geq$ 1418 psid, and
3)	RHR pump	$\geq$ 166 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.

Boron Injection Throttle Valves Valve Number	Safety Injection Throttle Valves Valve Number
NI-480	NI-488
NI-481	NI-489
NI-482	NI-490
NI-483	NI-491

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 325 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 423 gpm, and

b) The total pump flow rate is less than or equal to 675 gpm.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 4025 gpm.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the RHR 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 capable of injecting into the RCS whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. Two charging pumps may be operable and operating for ≤ 15 minutes to allow swapping charging pumps. Additional requirements are provided by Specification 3.4.9.3

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 centrifugal charging pumps and Safety Injection pumps, not capable of injecting into the RCS shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each pump has been isolated from the RCS by at least two isolation valves with power removed from the valve operators at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

3/4.5.4 (Deleted)

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A contained borated water volume of at least 372,100 gallons,

b. A boron concentration between the limits presented in the Core Operating Limits Report,

c. A minimum solution temperature of 70°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With the 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

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2) Verifying the boron concentration of the water.

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

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 or operator action during periods when containment isolation valves are open under administrative control,** and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions;

b. By verifying that each containment air lock is in compliance with 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 P_a, 14.8 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.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and the annulus 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.

**The following valves may be opened on an intermittent basis under administrative control: NC-141, NC-142, WE-13, WE-23, VX-34, VX-40, FW-11, FW-13, FW-4.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.30% by weight of the containment air per 24 hours at P_a, 14.8 psig, or

2) Less than or equal to L_t, 0.14% by weight of the containment air per 24 hours at a reduced pressure of P_t, 7.4 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 bypass leakage rate of less than 0.07 L_a for all penetrations identified as secondary containment bypass leakage paths when pressurized to P_a.

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

ACTION:

With (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) the combined bypass leakage rate exceeding 0.07 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.07 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972 or the mass-plot method:

SURVEILLANCE REQUIREMENTS (Continued)

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a, 14.8 psig, or at P_t, 7.4 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 either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1) Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a, or 0.25 L_t;

   2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at P_a, 14.8 psig, or P_t, 7.4 psig.

d. Type B and C tests shall be conducted with gas at P_a, 14.8 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks,

   2) Dual-ply bellows assemblies on containment penetrations between the containment building and the annulus, and

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

   4) Type C tests performed on containment penetrations M372, M373 without draining the glycol-water mixture from the seats of their diaphragm valves (NF-228A, NF-233B, and NF-234A), if meeting a zero indicated leakage rate (not including instrument error) for the diaphragm valves. These tests may be used in lieu of tests which are otherwise required by Section III.C.2(a) of 10 CFR 50, Appendix J to use air or nitrogen as

SURVEILLANCE REQUIREMENTS (Continued)

the test medium. The above required test pressure (P_a) and test interval are not changed by this exception.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.9.3 or 4.6.1.9.4, as applicable;

f. The combined bypass leakage rate shall be determined to be less than 0.07 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 14.8 psig, or P_t, 7.4 psig, during each Type A test;

g. Air locks shall be tested and demonstrated OPERABLE per Specification 4.6.1.3;

h. The space between each dual-ply bellows assembly on containment penetrations between the containment building and the annulus shall be vented to the annulus during Type A tests. Following completion of each Type A test, the space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3-5 psig to verify no detectable leakage or the dual-ply bellows assembly shall be subjected to a leak test with the pressure on the containment side of the dual-ply bellows assembly at P_a, 14.8 psig, or P_t, 7.4 psig, to verify the leakage to be within the limits of Specification 4.6.1.2f.;

i. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced Integrated Leakage Measurement System; and

j. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than 0.05 L_a at P_a, 14.8 psig.

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

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,

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

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of 14.8 psig,

b. By conducting overall air lock leakage tests at not less than P_a, 14.8 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, and

d. At least once per 6 months by conducting a pressure test to verify door seal integrity, with a measured leak rate of less than 15 standard cubic centimeters per minute.

#The provisions of Specification 4.0.2 are not applicable.

*This constitutes an exemption to Appendix J of 10 CFR 50.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained:

a. Between 75* and 100°F in the containment upper compartment, and

b. Between 100* and 120°F*** in the containment lower compartment.

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

ACTION:

With the containment average air temperature not conforming to the above limits, restore the air temperature to within the limits within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The primary containment upper compartment average air temperature shall be the weighted average** of ambient air temperature monitoring stations located in the upper compartment. Temperature readings will be obtained at least once per 24 hours from the elevation of 826 feet at the inlet of each upper containment ventilation unit.

4.6.1.5.2 The primary containment lower compartment average air temperature shall be the weighted average** of ambient air temperature monitoring stations located in the lower compartment. Temperature readings will be obtained at least once per 24 hours from the elevation of 745 feet at the inlet of each lower containment ventilation unit.

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*Lower limit may be reduced to 60°F in MODES 2, 3, and 4.

**The weighted average is the sum of each temperature multiplied by its respective containment volume fraction. In the event of inoperative temperature sensor(s), the weighted average shall be taken as the reduced total divided by one minus the volume fraction represented by the sensor(s) out of service.

***Containment lower compartment temperature may be between 120 and 125°F for up to 90 cumulative days per calendar year provided the lower compartment temperature average over the previous 365 days is less than 120°F.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73.

REACTOR BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the reactor building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

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

ACTION:

With the structural integrity of the reactor building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the reactor building shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the reactor building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the reactor building detected during the above required inspections shall be reported to the Commission pursuant to 10 CFR Sections 50.72, and 50.73.

CONTAINMENT SYSTEMS

ANNULUS VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 Two independent Annulus Ventilation Systems shall be OPERABLE.

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

ACTION:

a. With one Annulus Ventilation System inoperable for reasons other than the pre-heaters tested in 4.6.1.8.a and 4.6.1.8.d.5, 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.

b. With the pre-heaters tested in 4.6.1.8.a and 4.6.1.8.d.5 inoperable, restore the inoperable pre-heaters to OPERABLE status within 7 days, or file a Special Report in accordance with Specification 6.9.2 within 30 days specifying the reason for inoperability and the planned actions to return the pre-heaters to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.6.1.8 Each Annulus Ventilation System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS, by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the pre-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 ventilation system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance of 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 8000 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, and tested per ASTM D3803-89 has a methyl iodide penetration of less than 4%; and

   3) Verifying a system flow rate of 8000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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, and tested per ASTM D3803-89, has a methyl iodide penetration of less than 4%;

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 inches Water Gauge while operating the system at a flow rate of 8000 cfm ± 10%;

   2) Verifying that the system starts automatically on any Phase B Isolation test signal;

   3) Verifying that the filter cooling electric motor-operated bypass valves can be opened;

   4) Verifying that each system produces a negative pressure of greater than or equal to 0.5 inch W.G. in the annulus within 22 seconds after a start signal and that this negative pressure goes to -3.5 inches W.G. within 48 seconds after the start signal. Verifying that upon reaching a negative pressure of -3.5 inches W.G. in the annulus, the system switches into its recirculation mode of operation and that the time required for the annulus pressure to increase to -0.5 inch W.G. is greater than or equal to 278 seconds;

   5) Verifying that the pre-heaters dissipate 43.0 ± 6.4 kW at a nominal voltage of 600 VAC when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for DOP test aerosol while operating the system at a flow rate of 8000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 8000 cfm ± 10%.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 Each containment purge supply and/or exhaust isolation valve shall be OPERABLE and:

a. Each containment purge supply and/or exhaust isolation valve for the lower compartment (24-inch) and instrument room (12-inch and 24-inch) shall be sealed closed, and

b. The containment purge supply and/or exhaust isolation valve(s) for the upper compartment (24-inch) may be opened for up to 250 hours during a calendar year provided no more than one pair (one supply and one exhaust) are open at one time.

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

ACTION:

a. With any containment purge supply and/or exhaust isolation valve for the lower compartment or instrument room open or not sealed closed, close and/or seal closed that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment purge supply and/or exhaust isolation valve(s) for the upper compartment open for more than 250 hours during a calendar year, close any open 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.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.9.3 and/or 4.6.1.9.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 Each containment purge supply and/or exhaust isolation valve(s) for the lower compartment and instrument room shall be verified to be sealed closed at least once per 31 days.

4.6.1.9.2 The cumulative time that all containment purge supply and/or exhaust isolation valves for the upper compartment have been open during a calendar year shall be determined at least once per 7 days.

4.6.1.9.3 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed containment purge supply and/or exhaust isolation valve for the lower compartment and instrument room with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.

4.6.1.9.4 At least once per 3 months each containment purge supply and/or exhaust isolation valve for the upper compartment with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.01 L_a when pressurized to P_a.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2 Each Containment Spray System shall be demonstrated OPERABLE:

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

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 185 psig when tested pursuant to Specification 4.0.5;

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

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Phase B Isolation test signal,

   2) Verifying that each spray pump starts automatically on a Containment Phase B Isolation test signal,

   3) Verifying that the Containment Pressure Control System functions within the setpoint limits specified in Table 3.3-4, Item 6.

d. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE with isolation times less than or equal to the required isolation times.

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

ACTION:

With one or more containment isolation valve(s) 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.

e. The provisions of Specification 3.0.4 are not applicable provided that the affected penetration is isolated in accordance with ACTION b. or c. above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A Containment Isolation test signal, each Phase A isolation valve actuates to its isolation position,

b. Verifying that on a Phase B Containment Isolation test signal, each Phase B isolation valve actuates to its isolation position, and

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

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

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:

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.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using hydrogen gas mixtures to obtain calibration points of:

a. Zero volume percent hydrogen, and

b. Nine volume percent hydrogen.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent Containment Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a Recombiner System functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW, and

b. At least once per 18 months by:

   1) Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

   2) Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

   3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

HYDROGEN CONTROL DISTRIBUTED IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 Both trains of the Hydrogen Mitigation System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one train of the Hydrogen Mitigation System inoperable, restore the inoperable system to OPERABLE status within 7 days or decrease the surveillance interval of Specification 4.6.4.3.a from 92 days to 7 days on the OPERABLE train until the inoperable train is returned to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each train of the Hydrogen Mitigation System shall be demonstrated OPERABLE:

a. At least once per 92 days by energizing the supply breakers and verifying that at least 32 of 33 igniters are energized,* and

b. At least once per 18 months by verifying the temperature of each igniter is a minimum of 1700°F.

*Inoperable igniters must not be on corresponding redundant circuits which provide coverage for the same region.

3/4.6.5 ICE CONDENSER

ICE BED

LIMITING CONDITION FOR OPERATION

3.6.5.1 The ice bed shall be OPERABLE with:

a. The stored ice having a boron concentration of at least 1800 ppm boron as sodium tetraborate and a pH of 9.0 to 9.5,

b. Flow channels through the ice condenser,

c. A maximum ice bed temperature of less than or equal to 27°F,

d. A total ice weight of at least 2,099,790 pounds at a 95% level of confidence, and

e. 1944 ice baskets.

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

ACTION:

With the ice bed inoperable, restore the ice bed to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 The ice condenser shall be determined OPERABLE:

a. At least once per 12 hours by using the Ice Bed Temperature Monitoring System to verify that the maximum ice bed temperature is less than or equal to 27°F,

b. At least once per 9 months by:

1) Chemical analyses which verify that at least nine representative samples of stored ice have a boron concentration of at least 1800 ppm as sodium tetraborate and a pH of 9.0 to 9.5 at 20°C;

2) Weighing a representative sample of at least 144 ice baskets and verifying that each basket contains at least 1081 lbs of ice. The representative sample shall include 6 baskets from each of the 24 ice condenser bays and shall be constituted of

SURVEILLANCE REQUIREMENTS (Continued)

1 basket each from Radial Rows 1, 2, 4, 6, 8, and 9 (or from the same row of an adjacent bay if a basket from a designated row cannot be obtained for weighing) within each bay. If any basket is found to contain less than 1081 pounds of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The minimum average weight of ice from the 20 additional baskets and the discrepant basket shall not be less than 1081 pounds/basket at a 95% level of confidence.

The ice condenser shall also be subdivided into 3 groups of baskets, as follows: Group 1 - Bays 1 through 8, Group 2 - Bays 9 through 16, and Group 3 - Bays 17 through 24. The minimum average ice weight of the sample baskets from Radial Rows 1, 2, 4, 6, 8, and 9 in each group shall not be less than 1081 pounds/basket at a 95% level of confidence.

The minimum total ice condenser ice weight at a 95% level of confidence shall be calculated using all ice basket weights determined during this weighing program and shall not be less than 2,099,790 pounds; and

3) Verifying, by a visual inspection of at least two flow passages per ice condenser bay, that the accumulation of frost or ice on flow passages between ice baskets, past lattice frames, through the intermediate and top deck floor grating, or past the lower inlet plenum support structures and turning vanes is restricted to a thickness of less than or equal to 0.38 inch. If one flow passage per bay is found to have an accumulation of frost or ice with a thickness of greater than or equal to 0.38 inch, a representative sample of 20 additional flow passages from the same bay shall be visually inspected. If these additional flow passages are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one restricted flow passage per bay is evidence of abnormal degradation of the ice condenser.

c. At least once per 40 months by lifting and visually inspecting the accessible portions of at least two ice baskets from each one-third of the ice condenser and verifying that the ice baskets are free of detrimental structural wear, cracks, corrosion, or other damage. The ice baskets shall be raised at least 12 feet for this inspection.

ICE BED TEMPERATURE MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.2 The Ice Bed Temperature Monitoring System shall be OPERABLE with at least two OPERABLE RTD channels in the ice bed at each of three basic elevations (10'6", 30'9" and 55' above the floor of the ice condenser) for each one-third of the ice condenser.

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

ACTION:

a. With the Ice Bed Temperature Monitoring System inoperable, POWER OPERATION may continue for up to 30 days provided:

   1. The ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed;

   2. The last recorded mean ice bed temperature was less than or equal to 20°F and steady; and

   3. The ice condenser cooling system is OPERABLE with at least:

      a) 21 OPERABLE air handling units,

      b) 2 OPERABLE glycol circulating pumps, and

      c) 3 OPERABLE refrigerant units;

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

b. With the Ice Bed Temperature Monitoring System inoperable and with the Ice Condenser Cooling System not satisfying the minimum components OPERABILITY requirements of ACTION a.3. above, POWER OPERATION may continue for up to 6 days provided the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed and the last recorded mean ice bed temperature was less than or equal to 15°F and steady; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 The Ice Bed Temperature Monitoring System shall be determined OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours.

ICE CONDENSER DOORS

LIMITING CONDITION FOR OPERATION

3.6.5.3 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be closed and OPERABLE.

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

ACTION:

a. With one or more ice condenser doors open or otherwise inoperable (but capable of opening automatically), POWER OPERATION may continue for up to 14 days provided the ice bed temperature is monitored at least once per 4 hours and the maximum ice bed temperature is maintained less than or equal to 27°F; otherwise, restore the doors to their closed positions or OPERABLE status (as applicable) within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more ice condenser doors inoperable (not capable of opening automatically), restore all doors to OPERABLE status within 1 hour or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the following 30 hours

SURVEILLANCE REQUIREMENTS

4.6.5.3.1 Inlet Doors - Ice condenser inlet doors shall be:

a. Continuously monitored and determined closed by the inlet door position monitoring system, and

b. Demonstrated OPERABLE during shutdown at least once per 18 months by:

1) Verifying that the torque required to initially open each door is less than or equal to 675 inch pounds;

2) Verifying that each door is capable of opening automatically in that it is not impaired by ice, frost, debris, or other obstruction;

3) Testing each one of the doors and verifying that the torque required to open each door is less than 195 inch-pounds when the door is 40 degrees open. This torque is defined as the "door opening torque" and is equal to the nominal door torque plus a frictional torque component;

SURVEILLANCE REQUIREMENTS (Continued)

4) Testing each one of the doors and verifying that the torque required to keep each door from closing is greater than 78 inch-pounds when the door is 40 degrees open. This torque is defined as the "door closing torque" and is equal to the nominal door torque minus a frictional torque component; and

5) Calculation of the frictional torque of each door tested in accordance with 3) and 4), above. The calculated frictional torque shall be less than or equal to 40 inch-pounds.

4.6.5.3.2 Intermediate Deck Doors - Each ice condenser intermediate deck door shall be:

a. Verified closed and free of frost accumulation by a visual inspection at least once per 7 days, and

b. Demonstrated OPERABLE at least once per 3 months during the first year after the ice bed is fully loaded and at least once per 18 months thereafter by visually verifying no structural deterioration, by verifying free movement of the vent assemblies, and by ascertaining free movement when lifted with the applicable force shown below:

	Door	Lifting Force
1)	Adjacent to crane wall	Equal to or less than 37.4 lbs,
2)	Paired with door adjacent to crane wall	Equal to or less than 33.8 lbs,
3)	Adjacent to containment wall	Equal to or less than 31.8 lbs, and
4)	Paired with door adjacent to containment wall	Equal to or less than 31.0 lbs.

4.6.5.3.3 Top Deck Doors - Each ice condenser top deck door shall be determined closed and OPERABLE at least once per 92 days by visually verifying:

a. That the doors are in place, and

b. That no condensation, frost, or ice has formed on the doors or blankets which would restrict their lifting and opening if required.

INLET DOOR POSITION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 The Inlet Door Position Monitoring System shall be OPERABLE.

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

ACTION:

With the Inlet Door Position Monitoring System inoperable, POWER OPERATION may continue for up to 14 days, provided the Ice Bed Temperature Monitoring System is OPERABLE and the maximum ice bed temperature is less than or equal to 27°F when monitored at least once per 4 hours; otherwise, restore the Inlet Door Position Monitoring System to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 The Inlet Door Position Monitoring System shall be determined OPERABLE by:

a. Performing a CHANNEL CHECK at least once per 7 days and within 4 hours after receiving an "Ice Condenser Inlet Door Open" alarm on the control room annunciator portion of the system,

b. Performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 18 months, and

c. Verifying that the Monitoring System correctly indicates the status of each inlet door as the door is opened and reclosed during its testing per Specification 4.6.5.3.1.

DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

LIMITING CONDITION FOR OPERATION

3.6.5.5 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be OPERABLE and closed.

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

ACTION:

With a personnel access door or equipment hatch inoperable or open except for personnel transit entry, restore the door or hatch to OPERABLE status or to its closed position (as applicable) within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.5.1 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined closed by a visual inspection prior to increasing the Reactor Coolant System T_{avg} above 200°F and after each personnel transit entry when the Reactor Coolant System T_{avg} is above 200°F.

4.6.5.5.2 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined OPERABLE by visually inspecting the seals and sealing surfaces of these penetrations and verifying no detrimental misalignments, cracks or defects in the sealing surfaces, or apparent deterioration of the seal material:

a. Prior to final closure of the penetration each time it has been opened, and

b. At least once per 10 years for penetrations containing seals fabricated from resilient materials.

CONTAINMENT AIR RETURN AND HYDROGEN SKIMMER SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent Containment Air Return and Hydrogen Skimmer Systems shall be OPERABLE.

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

ACTION:

With one Containment Air Return and Hydrogen Skimmer System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6.1 Each Containment Air Return and Hydrogen Skimmer System shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by:

a. Verifying that the air return and hydrogen skimmer fans start automatically on a Containment Phase B Isolation (S_p) test signal after a 9 ± 1 minute delay and operate for at least 15 minutes;

b. Verifying that during air return fan operation with the air return fan damper closed and with the bypass dampers open, the fan motor current is less than or equal to 32 amps when the fan speed is 870 $\pm$ 30 rpm;

c. Verifying that with the hydrogen skimmer fan operating and the motor operated valve in its suction line closed, the fan motor current is less than or equal to 21.5 amps when the fan speed is 3599 $\pm$ 20 rpm;

d. Verifying that with the air return fan off, the motor operated damper in the air return fan discharge line to the containment's lower compartment opens automatically with a 10 $\pm$ 1 second delay after a Containment Phase B Isolation (S_p) test signal;

e. Verifying that with the air return fan operating, the check damper in the air return fan discharge line to the containment's lower compartment is open;

f. Verifying that the motor operated valve in the hydrogen skimmer suction line opens automatically and the hydrogen skimmer fans receive a start permissive signal; and

g. Verifying that with the fan off, the return air fan check damper is closed.

4.6.5.6.2 At least once per 18 months, each Containment Air Return and Hydrogen Skimmer System shall be demonstrated OPERABLE by verifying that the containment pressure control system functions within the setpoint limits specified in Table 3.3-4, Item 6.

FLOOR DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.7 The ice condenser floor drains shall be OPERABLE.

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

ACTION:

With the ice condenser floor drain inoperable, restore the floor drain to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.7 Each ice condenser floor drain shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

a. Verifying that valve gate opening is not impaired by ice, frost or debris,

b. Verifying that the valve seat is not damaged,

c. Verifying that the valve gate opens when a force of less than or equal to 66 lbs is applied, and

d. Verifying that the drain line from the ice condenser floor to the containment lower compartment is unrestricted.

REFUELING CANAL DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.8 The refueling canal drains shall be OPERABLE.

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

ACTION:

With a refueling canal drain inoperable, restore the drain to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.8 Each refueling canal drain shall be demonstrated OPERABLE.

a. Prior to increasing the Reactor Coolant System temperature above 200°F after each partial or complete filling of the canal with water by verifying that the valves in the drain lines are locked open and that the drain is not obstructed by debris, and

b. At least once per 92 days by verifying, through a visual inspection, that there is no debris that could obstruct the drain.

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

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

ACTION:

With the divider barrier seal inoperable, restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:

a. Removing two divider barrier seal test coupons and verifying that the physical properties of the test coupons are within the acceptable range of values shown in Table 3.6-3; and

b. Visually inspecting at least 95% of the seal's entire length and:

   1) Verifying that the seal and seal mounting bolts are properly installed, and

   2) Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

TABLE 3.6-3

DIVIDER BARRIER SEAL
ACCEPTABLE PHYSICAL PROPERTIES

MEMBRANE TYPE SEALS	TENSILE STRENGTH
MK 10	39.7 lbs
MK 11	39.7 lbs

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-3.

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 COLD SHUTDOWN within the following 30 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 COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5. Following testing, lift settings shall be within ± 1%.

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	(**)
2	(**)
3	(**)

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					LIFT SETTING(± 3%)***	ORIFICE SIZE
	Loop A	Loop B	Loop C	Loop D		
1.	SV 20	SV 14	SV 8	SV 2	1170 psig	12.174 in^2
2.	SV 21	SV 15	SV 9	SV 3	1190 psig	12.174 in^2
3.	SV 22	SV 16	SV 10	SV 4	1205 psig	16.00 in^2
4.	SV 23	SV 17	SV 11	SV 5	1220 psig	16.00 in^2
5.	SV 24	SV 18	SV 12	SV 6	1225 psig	16.00 in^2

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

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

***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 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) 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;

2) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

3) Verifying that the isolation valves in the auxiliary feedwater suction line from the upper surge tanks are open with power to the valve operators removed.

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*N.. applicable with steam pressure less than 900 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days on a STAGGERED BASIS by:

   1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 450 gpm; and

   2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 900 gpm when the secondary steam supply pressure is greater than 900 psig.* The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

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 upon receipt of an Auxiliary Feedwater Actuation test signal,

   2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal, and

   3) Verifying that the valve in the suction line of each auxiliary feedwater pump from the Nuclear Service Water System automatically actuates to its full open position within less than or equal to 13 seconds on a Low Suction Pressure test signal.

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* This verification is not required to be performed until 24 hours after achieving greater than or equal to 900 psig in the secondary side of the steam generator.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.3 The specific activity of the Secondary Coolant System shall be less than or equal to 0.10 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 0.10 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Specific Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross specific activity determination indicates concentrations greater than 10% of the allowable limit for radioiodine. b) Once per 6 months, whenever the gross specific activity determination indicates concentrations below 10% of the allowable limit for radioiodine.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.4 Each main steam line isolation valve (MSLIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1 - With one MSLIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within 2 hours.

MODES 2 AND 3 - With one MSLIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided:

a. The isolation valve is maintained closed, and

b. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.7.1.4 Each MSLIV shall be demonstrated OPERABLE by verifying full closure within 8 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

   2) Each component cooling water pump starts automatically on a Safety Injection and Station Blackout test signal.

3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With the unit specific portion of only one nuclear service water loop per unit OPERABLE, restore both unit specific loops to OPERABLE status within 72 hours or place the affected unit at least in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With only one of the shared portions as defined by Figure 3/4 7-1 of the Unit 1 and Unit 2 nuclear service water loops OPERABLE, restore the shared portion of the loops to OPERABLE status within 72 hours or place both units in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

   2) Each nuclear service water pump starts automatically on a Safety Injection and Station Blackout test signal.

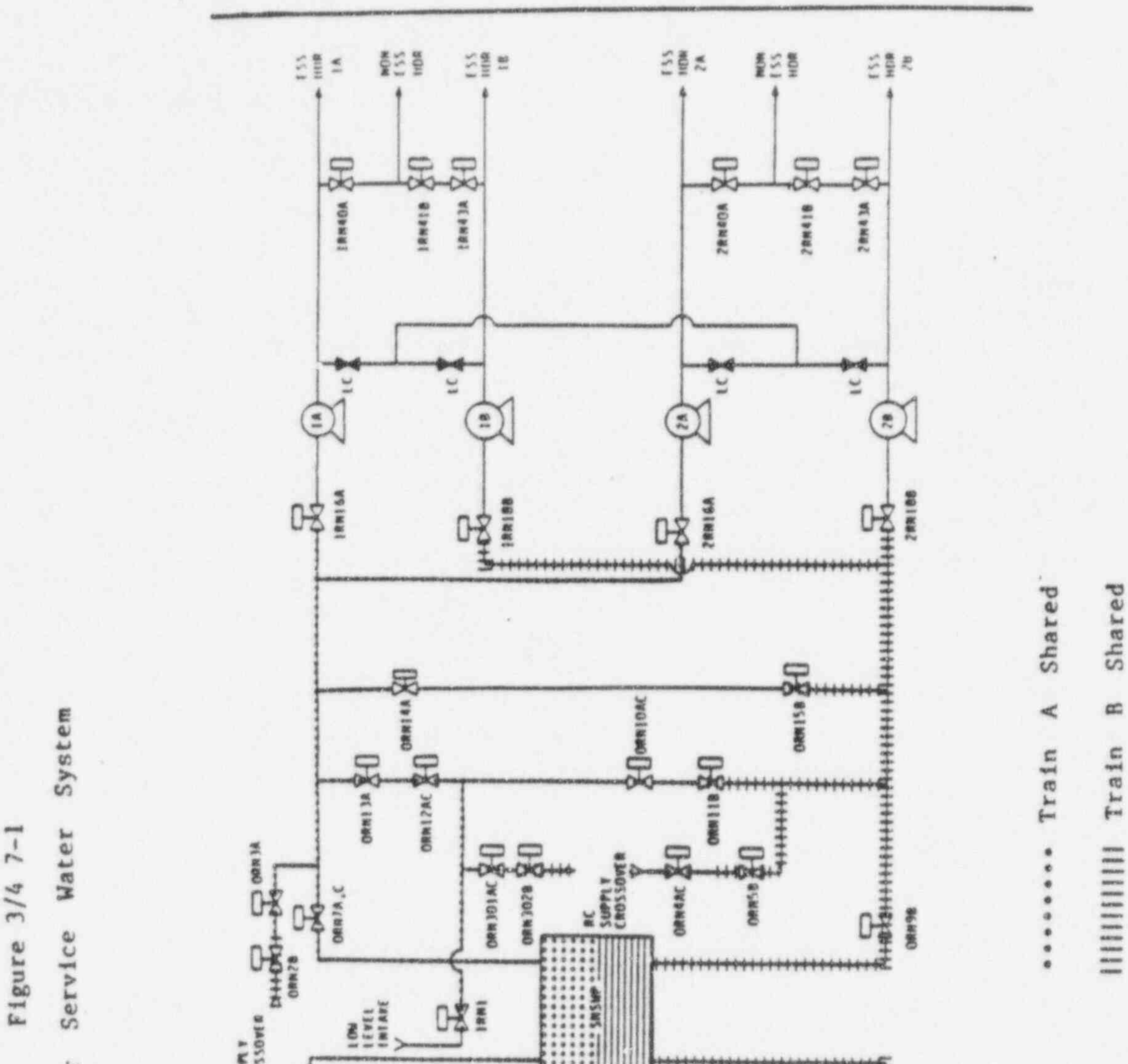

FIGURE 3/4 7-1

NUCLEAR SERVICE WATER SYSTEM

3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND

LIMITING CONDITION FOR OPERATION

3.7.5 The standby nuclear service water pond shall be OPERABLE with:

a. A minimum water level at or above elevation 739.5 feet Mean Sea Level, USGS datum, and

b. An average water temperature of less than or equal to 82°F at elevation 722 feet.

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

ACTION: (Units 1 and 2)

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The standby nuclear service water pond shall be determined OPERABLE:

a. At least once per 24 hours by verifying the water level to be within its limit,

b. At least once per 24 hours during the months of July, August and September by verifying the water temperature to be within its limit, and

c. At least once per 12 months by visually inspecting the dam and verifying no abnormal degradation, erosion, or excessive seepage.

3/4.7.6 CONTROL AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Area Ventilation Systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION: (Units 1 and 2)

MODES 1, 2, 3 and 4:

a. With one Control Area Ventilation System inoperable for reasons other than the heaters specified in 4.7.6.b and 4.7.6.e.4, 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.

b. With the heaters tested in 4.7.6.b and 4.7.6.e.4 inoperable, restore the inoperable heaters to OPERABLE status within 7 days, or file a Special Report in accordance with Specification 6.9.2 within 30 days, specifying the reason for inoperability and the planned actions to return the heaters to OPERABLE status.

MODES 5 and 6:

a. With one Control Area Ventilation System inoperable for reasons other than the heaters specified in 4.7.6.b and 4.7.6.e.4, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Area Ventilation System in the recirculation mode; and

b. With both Control Area Ventilation Systems inoperable for reasons other than the heaters specified in 4.7.6.b and 4.7.6.e.4, or with the OPERABLE Control Area Ventilation 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.

c. The provisions of Specification 3.0.4 are not applicable.

d. With the heaters tested in 4.7.6.b and 4.7.6.e.4 inoperable, restore the inoperable heaters to OPERABLE status within 7 days, or file a Special Report in accordance with Specification 6.9.2 within 30 days, specifying the reason for inoperability and the planned actions to return the heaters to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Area Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours, by verifying that the control room air temperature is less than or equal to 90°F;

b. At least once per 31 days on a STAGGERED TEST BASIS, by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters operating;

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 of 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 2000 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, and tested per ASTM D3803-89, has a methyl iodide penetration of less than 0.95%; and

   3) Verifying a system flow rate of 2000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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, and tested per ASTM D3803-89, has a methyl iodide penetration of less than 0.95%;

e. At least once per 18 months, by:

   1) Verifying that the pressure drop across the combined prefilters, HEPA filters and charcoal adsorber banks is less than 5 inches Water Gauge while operating the system at a flow rate of 2000 cfm ± 10%;

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that upon actuation of a diesel generator sequencer the system automatically switches into a mode of operation with flow through the HEPA filters and charcoal adsorber banks;

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch W.G. relative to the outside atmosphere during system operation; and

4) Verifying that the heaters dissipate 10 ± 1.0 kW at a nominal voltage of 600 VAC when tested in accordance with ANSI N510-1975.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 2000 cfm ± 10%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 2000 cfm ± 10%.

3/4.7.7 AUXILIARY BUILDING FILTERED VENTILATION EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 The Unit 1 and Unit 2 Auxiliary Building Filtered Ventilation Exhaust Systems shall be OPERABLE.

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

ACTION: (Units 1 and 2)

a. With one Auxiliary Building Filtered Ventilation Exhaust System filter package inoperable, restore the inoperable filter to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one Auxiliary Building Filtered Ventilation Exhaust System flowpath inoperable (except carbon and HEPA filter package components and except as addressed by c.1 and c.2 below) restore the inoperable flowpath 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.

c.1 With one Auxiliary Building Filtered Ventilation Exhaust System able to maintain a negative pressure but unable to maintain 0.125" W.G. at the ECCS pump room relative to outside atmosphere, restore system ability to maintain 0.125" W.G. within the next 7 days or be in a least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c.2 With one Auxiliary Building Filtered Ventilation Exhaust System unable to maintain a negative pressure at the ECCS pump room relative to outside atmosphere, restore system ability to maintain a negative pressure within the next 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With both Unit 1 and Unit 2 Auxiliary Building Filtered Ventilation Exhaust Systems inoperable, restore at least one inoperable system to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each unit's Auxiliary Building Filtered Ventilation Exhaust System filter package shall be demonstrated OPERABLE:

a. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or carbon adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

(1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance of 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 45,700 cfm ± 10% (both fans operating);

(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 an acceptance criteria for methyl iodide penetration of less than 10% at 30°C test temperature, and

b. After every 1440 hours of carbon adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets an acceptance criteria for methyl iodide penetration of less than 10% at 30°C test temperature, and

c. At least once per 18 months, by verifying that the pressure drop across the combined HEPA filters and carbon adsorber banks of less than 6 inches Water Gauge while operating the system at a flow rate of 45,700 cfm ± 10% (both fans operating), and

d. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 45,700 cfm ± 10% (both fans operating); and

e. After each complete or partial replacement of a carbon adsorber bank, by verifying that the carbon adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 45,700 cfm ± 10% (both fans operating).

4.7.7.2 Each unit's Auxiliary Building Filtered Ventilation Exhaust System flowpath shall be demonstrated OPERABLE:

a. At least once per 31 days, by initiating, from the control room, flow through the HEPA filters and carbon adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or carbon adsorber housings, or (2) following

SURVEILLANCE REQUIREMENTS (Continued)

painting, fire, or chemical release in any ventilation zone communicating with the system, by verifying a system flow rate of 45,700 cfm ± 10% (both fans operating) during system operation when tested in accordance with ANSI N510-1980.

c. At least once per 18 months, by verifying that the system starts on a Safety Injection test signal and directs its exhaust flow through the HEPA filters and carbon adsorbers.

4.7.7.3 Each unit's Auxiliary Building Filtered Ventilation Exhaust System shall be demonstrated OPERABLE, at least once per 18 months, by verifying that the system maintains the ECCS pump room at a negative pressure relative to outside atmosphere.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on non-safety-related systems and then only if the failure or the failure of the system on which they are installed would not have an adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4.
MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program 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 and may be treated independently. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each category of snubber shall be determined based upon the criteria provided in Table 4.7-2 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment No. 126.

TABLE 4.7-2

SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Notes 1 and 2)	Column A Extended Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. The categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category size and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as described by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

TABLE 4.7-2 (continued)

SNUBBER VISUAL INSPECTION INTERVAL

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

SURVEILLANCE REQUIREMENTS (continued)

c. Refueling Outage Inspections

At each refueling, the systems which have the potential for a severe dynamic event, specifically, the main steam system (upstream of the main steam isolation valves) the main steam safety and power-operated relief valves and piping, auxiliary feedwater system, main steam supply to the auxiliary feedwater pump turbine, and the letdown and charging portion of the CVCS system shall be inspected to determine if there has been a severe dynamic event. In case of a severe dynamic event, mechanical snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the mechanical snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying freedom of motion using one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; (3) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced or repaired before returning to power. The requirements of Specification 4.7.8b. are independent of the requirements of this specification.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify: (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections "shall be classified as unacceptable and may be reclassified acceptable" 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 4.7.8f. A hydraulic snubber found with the fluid port uncovered and all hydraulic snubbers found connected to an inoperable common reservoir shall be classified as unacceptable and may be reclassified acceptable by functionally testing each snubber starting with the piston in the as-found setting, extending the piston rod in the tension direction.

e. Functional Tests

During the first refueling shutdown and at least once per refueling thereafter, a representative sample of snubbers shall be tested using one of the following sample plans. The large bore steam generator hydraulic snubbers shall be treated as a separate population for functional test purposes. A 10% random sample from previously untested snubbers shall be tested at least once per refueling outage until the entire population has been tested. This testing cycle shall then begin anew. For each large bore steam

e. Functional Tests (Continued)

generator hydraulic snubber that does not meet the functional test acceptance criteria, at least 10% of the remaining population of untested snubbers for that testing cycle shall be tested. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC shall be notified of the sample plan selected prior to the test period.

1) At least 10% of the snubbers required by Specification 3.7.8 shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.8f., an additional 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested; or

2) A representative sample of the snubbers required by Specification 3.7.8 shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers found not meeting the acceptance requirements of Specification 4.7.8f (failures). The cumulative number of snubbers tested is denoted by "N." Test results shall be plotted sequentially in the order of sample assignment (i.e., each snubber shall be plotted by its order in the random sample assignments, not by the order of testing). If at any time the point plotted falls in the "Accept" region, testing of snubbers may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers shall be tested until the point falls in the "Accept" region, or all the snubbers required by Specification 3.7.8 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, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of fifty-five (55) snubbers shall be functionally tested. For each snubber 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. This can be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber should be plotted as soon as it is tested. If the point plotted falls on or below the "Accept" line, testing may be discontinued. If the point plotted falls above the "Accept" line, testing must continue unless all snubbers have been tested.

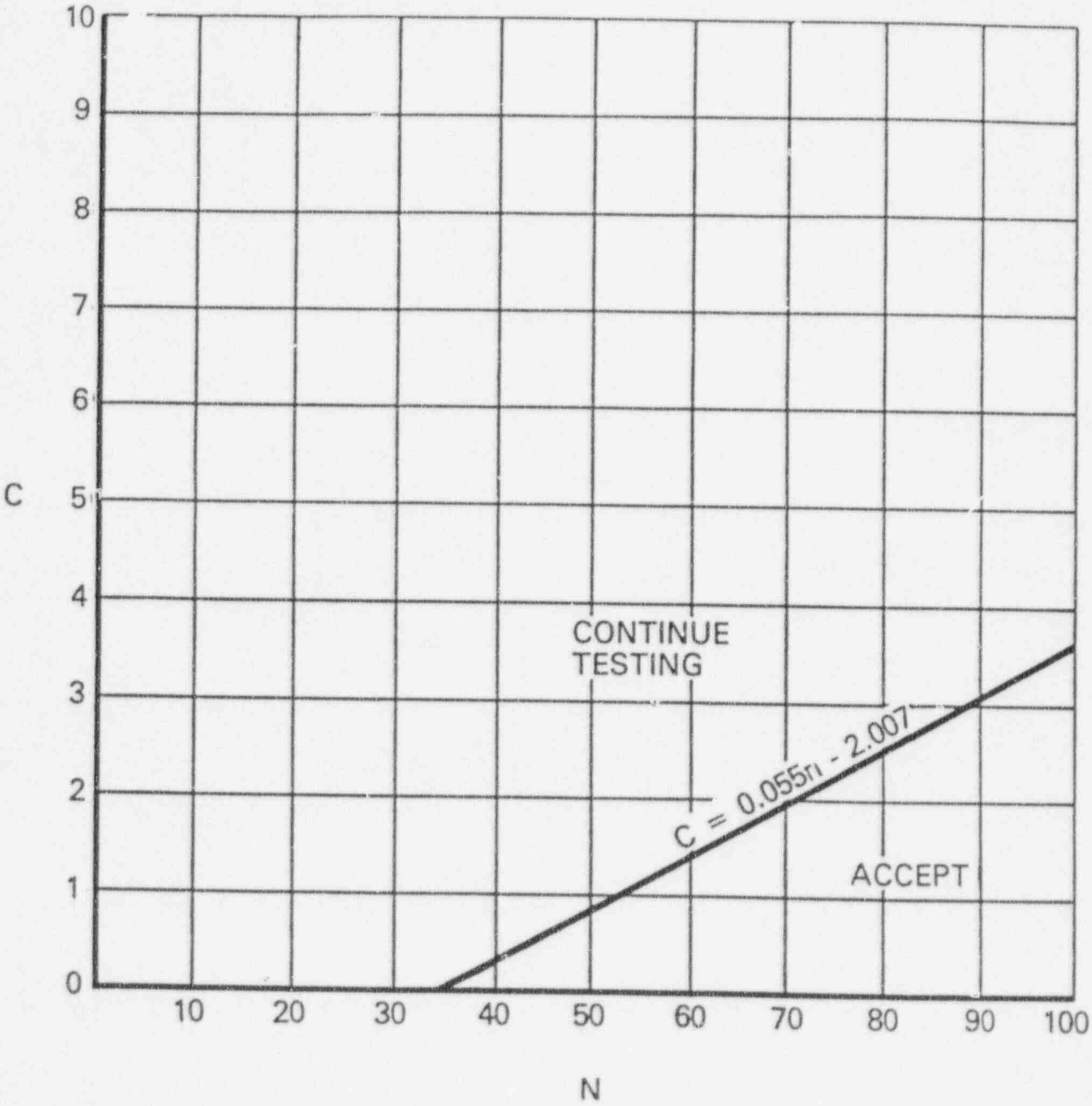

FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

e. Functional Tests (Continued)

The representative samples for the functional test sample plans shall be randomly selected from the snubbers required by Specification 3.7.8 and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of sizes, and capacities. Snubbers placed in the same locations 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 testing results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, 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 activate 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 evaluated in a manner to ensure their OPERABILITY. This testing requirement shall be independent of the requirements stated in Specification 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result 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 freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the expected service life will not be exceeded during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.3 Reports - A report shall be prepared 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.

3/4.7.10 DELETED

3/4.7.11 DELETED

3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature of each area shown in Table 3.7-6 shall be maintained within the limits indicated in Table 3.7-6.

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. For more than 8 hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 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. Containment Spray Pump Rooms -	145
2. Miscellaneous Terminal Cabinets	
a. TB208-209 (Turbine Bldg.)	150
b. TB496 (Fuel Bldg.)	150
3. Residual Heat Removal Pump Rooms	145
4. Diesel Generator Rooms	125
5. Spent Fuel Pool Cooling Pump Room	145

3/4.7.13 GROUNDWATER LEVEL

LIMITING CONDITION FOR OPERATION

3.7.13 The groundwater level shall be maintained at elevations less than the values in Table 3.7-7 for the five (5) Auxiliary Building monitors listed in Table 3.7-7.

APPLICABILITY: At all times.

ACTION: For Units 1 and 2.

If groundwater level for any three (3) of the five (5) monitors is above the values shown in Table 3.7-7, take the following actions:

1. Within one hour, reduce the groundwater level to below the values shown in Table 3.7-7; or,
2. Be in at least HOT STANDBY within 6 hours, and HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.13.1 During each shift, the groundwater level shall be demonstrated to be within the values of Table 3.7-7 by the absence of alarms or by visual observation of the monitor level gauge.

4.7.13.2 Each groundwater level monitor instrument/loop for locations listed in Table 3.7-7 shall be demonstrated OPERABLE at least once per year by the performance of a loop calibration and operational test.

TABLE 3.7-7

AUXILIARY BUILDING GROUNDWATER LEVEL MONITORS

LOCATION	INTERIOR/ EXTERIOR ELEVATION	UNIT
	(Feet - Mean Sea Level)	
PP-51	Interior 731' - 0"	1
QQ-56	Interior 731' - 0"	1 & 2
PP-61	Interior 731' - 0"	2
West Wall	Exterior 731' - 0"	1
East Wall	Exterior 731' - 0"	2

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the Onsite Essential Auxiliary Power System, and

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

   1) A separate day tank containing a minimum volume of 120 gallons of fuel,
   2) A separate Fuel Storage System containing a minimum volume of 39,500 gallons of fuel,
   3) A separate fuel transfer pump.

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

ACTION:

a. With an offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; separately demonstrate the operability of two diesel generators by performing Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5) within 24 hours unless this surveillance was performed within the previous 24 hours, or unless the diesel is operating, 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. source by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; demonstrate the operability of the remaining diesel generator by performing Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5) within 8 hours unless this surveillance was performed within the previous 24 hours, or unless the diesel is operating**; 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; with the

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*A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5).

**This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

ACTION: (Continued)

diesel generator restored to OPERABLE status, follow action statement a; with the offsite circuit restored to OPERABLE status, follow action statement d.

c. With one diesel generator inoperable in addition to ACTION b. or d. 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

   2. When in MODE 1, 2, or 3 with a steam pressure greater than 900 psig, the steam-driven auxiliary feedwater pump is OPERABLE.

   If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With a diesel generator of the above required A.C. electrical power sources inoperable*, demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter; and unless the inoperability of the diesel was due to preplanned testing or maintenance, demonstrate the OPERABILITY of the remaining diesel generator by performing Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5) within 24 hours or unless the diesel is operating**, restore 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.

e. With two of the above required offsite A.C. circuits inoperable, separately demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5) within 8 hours, 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 and in COLD SHUTDOWN within the following 30 hours. With only one offsite source restored, follow action statement a.

f. With two of the above required diesel generators inoperable*, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours

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*A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirements 4.8.1.1.2a.4) and 4.8.1.1.2a.5).

**This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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. With one diesel generator restored, follow action statement d.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Essential Auxiliary Power System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the day tank,

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least 488 rpm in less than or equal to 11 seconds.* The generator voltage and frequency shall be at least 4160 volts and 57 Hz within 11 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

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*The diesel generator start (11 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded to greater than or equal to 3000 kW in less than or equal to 60 seconds, and to 4000 kW within 10 minutes and operates for at least 60 minutes, and

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By removing accumulated water:

1) From the day tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and

2) From the storage tank at least once per 31 days.

c. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the 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:

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 at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2) By verifying within 31 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.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks 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.

e. At least once per 18 months, by:

   1) Subjecting the diesel to an inspection, during shutdown, in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;

   2) Verifying, during shutdown, the generator capability to reject a load of greater than or equal to 576 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz;

   3) Verifying, during shutdown, the generator capability to reject a load of 4000 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, during shutdown, and:

      a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

      b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected blackout loads through the load sequencer and operates for greater than or equal to 5 minutes while the generator is loaded with the blackout loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

   5) Verifying that on an ESF actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be at least 4160 volts and 57 Hz within 11 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within 4160 ± 420 volts and 60 ± 1.2 Hz during this test;

   6) Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

      a) Verifying, during shutdown, deenergization of the emergency busses and load shedding from the emergency busses;

SURVEILLANCE REQUIREMENTS (Continued)

b) Verifying, during shutdown, the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

c) Verifying, during shutdown,* that all automatic diesel generator trips, except engine overspeed, lube oil pressure, generator time overcurrent, and generator differential are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) [Deleted, Left Blank]

8) Verifying, during shutdown, the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded between 4200 kW and 4400 kW** and during the remaining 22 hours of this test, the diesel generator shall be loaded between 3800 kW and 4000 kW.** The generator voltage and frequency shall be at least 4160 volts and 57 Hz within 11 seconds after the start signal. The steady-state generator voltage and frequency shall be maintained within 4160 ± 420 volts and 60 ± 1.2 Hz during this test. Within 5 minutes of shutting down the diesel generator, restart the diesel generator and verify that the generator voltage and frequency reaches at least 4160 volts and 57 Hz within 11 seconds.***

9) Verifying that the auto-connected loads to each diesel generator do not exceed the 2-hour rating of 4400 kW;

10) Verifying, during shutdown, the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
b) Transfer its loads to the offsite power source, and
c) Be restored to its standby status.

11) Verifying, during shutdown, that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

SURVEILLANCE REQUIREMENTS (Continued)

12) 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;

13) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block are within the tolerances shown in Table 4.8-2;

14) Verifying, during shutdown,* that the following diesel generator lockout features prevent diesel generator starting only when required:

a) Turning gear engaged, or
b) Emergency stop.

15) Verifying, during shutdown, that with all diesel generator air start receivers pressurized to less than or equal to 220 psig and the compressors secured, the diesel generator starts at least 2 times from ambient conditions and accelerates to at least 488 rpm in less than or equal to 11 seconds.

f. 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 488 rpm in less than or equal to 11 seconds; and

g. 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

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*This Surveillance Requirement may be performed in conjunction with periodic preplanned preventative maintenance activity that causes the diesel generator to be inoperable provided that performance of the surveillance requirement does not increase the time the diesel generator would be inoperable for the PM activity alone.

**Diesel generator loadings for the purpose of this surveillance may be in accordance with vendor recommendations. The purpose of the load range is to prevent overloading the engine and momentary excursions outside of the range shall not invalidate the test.

***If there is a test failure during the 24-hour test run, the hot restart test can be performed prior to completing the 24-hour test provided the diesel generator had operated for at least 2 hours loaded between 3800 and 4000 kW.**

SURVEILLANCE REQUIREMENTS (Continued)

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 non-valid, shall be reported to the Commission pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

4.8.1.1.4 Diesel Generator Batteries - Each diesel generator 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

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

   2) The overall battery voltage is greater than or equal to 125 volts under a float charge.

b. At least once per 18 months by verifying that:

   1) The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration;

   2) The battery-to-battery and terminal connections are clear, tight, free of corrosion and coated with anti-corrosion material; and

   3) The battery capacity is adequate to supply and maintain in OPERABLE status its emergency loads when subjected to a battery service test.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS *	NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
≤ 1	≤ 4	Once per 31 days
≥ 2**	≥ 5	Once per 7 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, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new conditions is completed, provided that the overhaul including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer and if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine surveillance requirements of Specifications 4.8.1.1.2.a.4) and 4.8.1.1.2.a.5); the remaining four tests in accordance with the 184-day requirements specified in the footnote to Specification 4.8.1.1.2.a.4) and Specification 4.8.1.1.2.a.5). If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one.

TABLE 4.8-2

LOAD SEQUENCING TIMES

LOAD GROUP NUMBER	SEQUENCE TIME (Seconds)
Initiate Timer (T_o)	9.7 ± 0.3
1 (T_1)	T_o + 0.9 ± 0.1
2 (T_2)	T_o + 5.6 ± 0.4
3 (T_3)	T_o + 9.4 ± 0.6
4 (T_4)	T_o + 14.1 ± 0.9
5 (T_5)	T_o + 18.4 ± 1.2
6 (T_6)	T_o + 23.1 ± 1.4
7 (T_7)	T_o + 28.3 ± 1.7
8 (T_8)	T_o +530.0 ± 60.0
9 (T_9)	T_8 + 56.0 ± 4.0
10 (T_{10})	T_8 +112.3 ± 7.0

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One circuit between the offsite transmission network and the Onsite Essential Auxiliary Power System, and

b. One diesel generator with:

1) A day tank containing a minimum volume of 120 gallons of fuel,

2) A Fuel Storage System containing a minimum volume of 28,000 gallons of fuel, and

3) A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 4.5 square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5), 4.8.1.1.3, and 4.8.1.1.4.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following D.C. channels shall be OPERABLE and energized:

a. Channel 1 consisting of 125-Volt D.C. Bus No. EVDA, 125-Volt D.C. Battery Bank No. EVCA and a full-capacity charger,*#

b. Channel 2 consisting of 125-Volt D.C. Bus No. EVDB, 125-Volt D.C. Battery Bank No. EVCB and a full-capacity charger,*#

c. Channel 3 consisting of 125-Volt D.C. Bus No. EVDC, 125-Volt D.C. Battery Bank No. EVCC and a full-capacity charger,*# and

d. Channel 4 consisting of 125-Volt D.C. Bus No. EVDD, 125-Volt D.C. Battery Bank No. EVCD and a full-capacity charger,*#

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

ACTION: (Units 1 and 2)

a. With one 125-volt D.C. bus inoperable or not energized, restore the inoperable bus to OPERABLE and energized status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125-volt D.C. battery and/or its normal and standby chargers inoperable or not energized, either:

   1. Restore the inoperable battery and/or charger to OPERABLE and energized 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, or

   2. Energize the associated bus with an OPERABLE battery bank via OPERABLE tie breakers within 2 hours; operation may then continue for up to 72 hours from time of initial loss of OPERABILITY, otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*A vital bus may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided the vital busses associated with the other battery banks are OPERABLE and energized.

#During periods of station modification associated with battery, main and tie breaker replacement only, the loads of a DC bus may be energized from a same train DC bus via temporary cables and breakers connecting to the same train DC bus directly and bypassing the de-energized DC bus. A one time allowable outage time up to 112 hours is granted for each DC bus, one at a time, to allow for replacement of these breakers. Footnote * shall not be applied to any of the busses during the 112 hour period.

SURVEILLANCE REQUIREMENTS

4.8.2.1.1 Each D.C. channel shall be determined OPERABLE and energized with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment, indicated power availability from the charger and battery, and voltage on the bus of greater than or equal to 125 volts.

4.8.2.1.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days:

1) Verifying that the parameters in Table 4.8-3 meet the Category A limits, and

2) Verifying total battery terminal voltage is greater than or equal to 125 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge (battery terminal voltage below 110 volts), or battery overcharge (battery terminal voltage above 150 volts), by:

1) Verifying that the parameters in Table 4.8-3 meet the Category B limits,

2) Verifying there is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3) Verifying that the average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1) The cells, cell plates (if visible), and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms; and

4) The battery charger will supply at least 400 amperes at a minimum of 125 volts for at least 1 hour.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by verifying that the battery capacity is adequate to either:

   1) Supply and maintain in OPERABLE status all of the actual emergency loads for 1 hour when the battery is subjected to a battery service test, or

   2) Supply a dummy load of greater than or equal to 440 amperes for 60 minutes while maintaining the battery terminal voltage greater than or equal to 105 volts.

e. At least once per 60 months by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval, this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1.2d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 80% of the manufacturer's rating.

TABLE 4.8-3

BATTERY SURVEILLANCE REQUIREMENTS (Gould Cells)

	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 (c)	> 2.07 volts
Specific Gravity (a)	≥ 1.200 (b)	≥ 1.195 Average of all connected cells > 1.205	Not more than .020 below the average of all connected cells or ≥ 1.195 Average of all connected cells ≥ 1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps when on charge.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category 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.

TABLE 4.8-3 (Continued)

BATTERY SURVEILLANCE REQUIREMENTS (AT&T Cells)

	Category A (1)		Category B(2)	Category C(3)
Parameter	Limits for each designated pilot cell		Limits for each connected cell	Allowable value for each connected cell
Electrolyte Level	≥ Minimum level indication mark, and ≤ 1/4" above maximum level indication mark		≥ Minimum level indication mark, and ≤ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.20 Volts		≥ 2.17 Volts (4)	> 2.14 Volts
Specific (5) Gravity	≥ 1.285 (6)	CELL	≥ 1.280	Not more than 0.020 below the average of all connected cells or ≥ 1.280
		BATTERY	Average of all connected cells > 1.285 (7)	Average of all connected cells ≥ 1.280 (6)(7)

(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 C measurements are taken and found to be within their allowable values. All Category B parameter(s) must be within limits in 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 C parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category C parameter not within its allowable value indicates an INOPERABLE battery.
(4) Corrected for average electrolyte temperature.
(5) Corrected for electrolyte temperature and level.
(6) Or battery charging current is less than 2 amps when on float charge.
(7) With no more than 5 cells at the minimum limits.

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following D.C. electrical equipment and bus shall be OPERABLE and energized:

a. Two - 125-volt D.C. busses, and

b. Two - 125-volt D.C. battery banks and chargers associated with the above D.C. busses.

APPLICABILITY: MODES 5 and 6.

ACTION: (Units 1 and 2)

With less than the above complement of D.C. equipment and bus OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2.1 The above required 125-volt D.C. busses shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the bus.

4.8.2.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Specification 4.8.2.1.2.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following A.C. electrical busses and inverters shall be OPERABLE and energized with tie breakers open both between redundant busses within the unit and between the two units:

a. 4160-Volt Emergency Bus ETA,
b. 4160-Volt Emergency Bus ETB,
c. 600-Volt Emergency Bus ELXA,
d. 600-Volt Emergency Bus ELXB,
e. 600-Volt Emergency Bus ELXC,
f. 600-Volt Emergency Bus ELXD,
g. 120-Volt A.C. Vital Bus EKVA energized from Inverter EVIA connected to D.C. Channel 1,*#
h. 120-Volt A.C. Vital Bus EKVB energized from Inverter EVIB connected to D.C Channel 2,*#
i. 120-Volt A.C. Vital Bus EKVC energized from Inverter EVIC connected to D.C. Channel 3,*# and
j. 120-Volt A.C. Vital Bus EKVD energized from Inverter EVID connected to D.C. Channel 4.*#

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

ACTION:

a. With less than the above complement of A.C. busses OPERABLE and energized, restore the inoperable busses to OPERABLE and energized status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
b. With one inverter inoperable, energize the associated A.C. Vital Bus within 8 hours; restore the inoperable inverter to OPERABLE and energized status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified A.C. busses and inverters shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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*An inverter may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided: (1) its vital bus is OPERABLE and energized, and (2) the vital busses associated with the other battery banks are OPERABLE and energized. An inverter may be disconnected from its D.C. source for up to 72 hours provided the conditions of ACTION b. of Specification 3.8.2.1 are satisfied.

#During periods of station modification associated with battery, main and tie breaker replacement only, two channel related inverters maybe energized from a same train DC bus via temporary cables and breakers connecting to the same train DC bus directly and bypassing the associated de-energized DC bus. A one time allowable outage time up to 112 hours is granted for each DC bus, one at a time, to allow for replacement of these breakers. Footnote * shall not be applied to any of the busses during the 112 hour period.

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following A.C. electrical busses and inverters shall be OPERABLE and energized:

a. One - 4160-volt emergency bus,

b. Two - 600-volt emergency busses in a single train, and

*c. Two - 120-volt A.C. vital busses energized from their respective inverters connected to their respective D.C. channels.

APPLICABILITY MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours. Specification 3.8.3.2c. applies to both Units 1 and 2.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified A.C. busses and inverters shall be determined energized in the required manner at least once per 7 days by verifying correct alignment and indicated voltage on the busses.

*Required for both Units 1 and 2.

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 shown in FSAR Chapter 16 shall be OPERABLE.

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

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) shown in FSAR Chapter 16 inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices shown in FSAR Chapter 16 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 as specified in FSAR Chapter 16, and

SURVEILLANCE REQUIREMENTS (Continued)

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. For the lower voltage circuit breakers the nominal Trip Setpoint and overcurrent response times are listed in FSAR Chapter 16. 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; and

3) A fuse inspection and maintenance program will be maintained to ensure that:

1. The proper size and type of fuse is installed,

2. The fuse shows no sign of deterioration, and

3. The fuse connections are tight and clean.

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.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to the minimum boron concentration specified in the Core Operating Limits Report.

APPLICABILITY: MODE 6*, with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to the minimum boron concentration specified in the Core Operating Limits Report, 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 NV-250 shall be verified closed under administrative control at least once per 72 hours; or, NV-131, NV-140, NV-176, NV-468, NV-808, and either NV-132 or NV-1026 shall be verified closed under administrative control at least once per 12 hours when necessary to makeup to the RWST during refueling operations.

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*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE and operating with Alarm Setpoints at 0.5 decade above steady-state count rate, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 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 hatch closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Exhausting through OPERABLE Reactor Building Containment Purge Exhaust System HEPA filters and charcoal adsorbers.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment building penetrations shall be determined to be in its required condition 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.

4.9.4.2 The Reactor Building Containment Purge Exhaust System shall be demonstrated OPERABLE:

a. 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 system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance of 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 21,000 cfm ± 10% (both exhaust units operating);

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

3) Verifying a system flow rate of 21,000 cfm ± 10% (both exhaust units operating) during system operation when tested in accordance with ANSI N510-1975.

b. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

c. At least once per 18 months, by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 21,000 cfm ± 10% (both exhaust units operating);

d. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 21,000 cfm ± 10% (both exhaust units operating); and

e. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 21,000 cfm ± 10% (both exhaust units operating).

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1) A minimum capacity of 3250 pounds, and

   2) An overload cutoff limit less than or equal to 2900 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

   1) A minimum capacity of 610 pounds, and

   2) A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cutoff when the crane load exceeds 2900 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 610 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 3000 pounds* shall be prohibited from travel over fuel assemblies in the storage pool. Truck casks shall be carried along the path outlined in Figure 3.9-1 in the fuel pit and fuel pool area.

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 Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The weight of each load, other than a fuel assembly and control rod, shall be verified to be less than 3000 pounds prior to moving it over fuel assemblies.*

*Weir gates of the spent fuel pool may be moved by crane over the stored fuel provided the spent fuel has decayed for at least 17.5 days since last being part of a core at power.

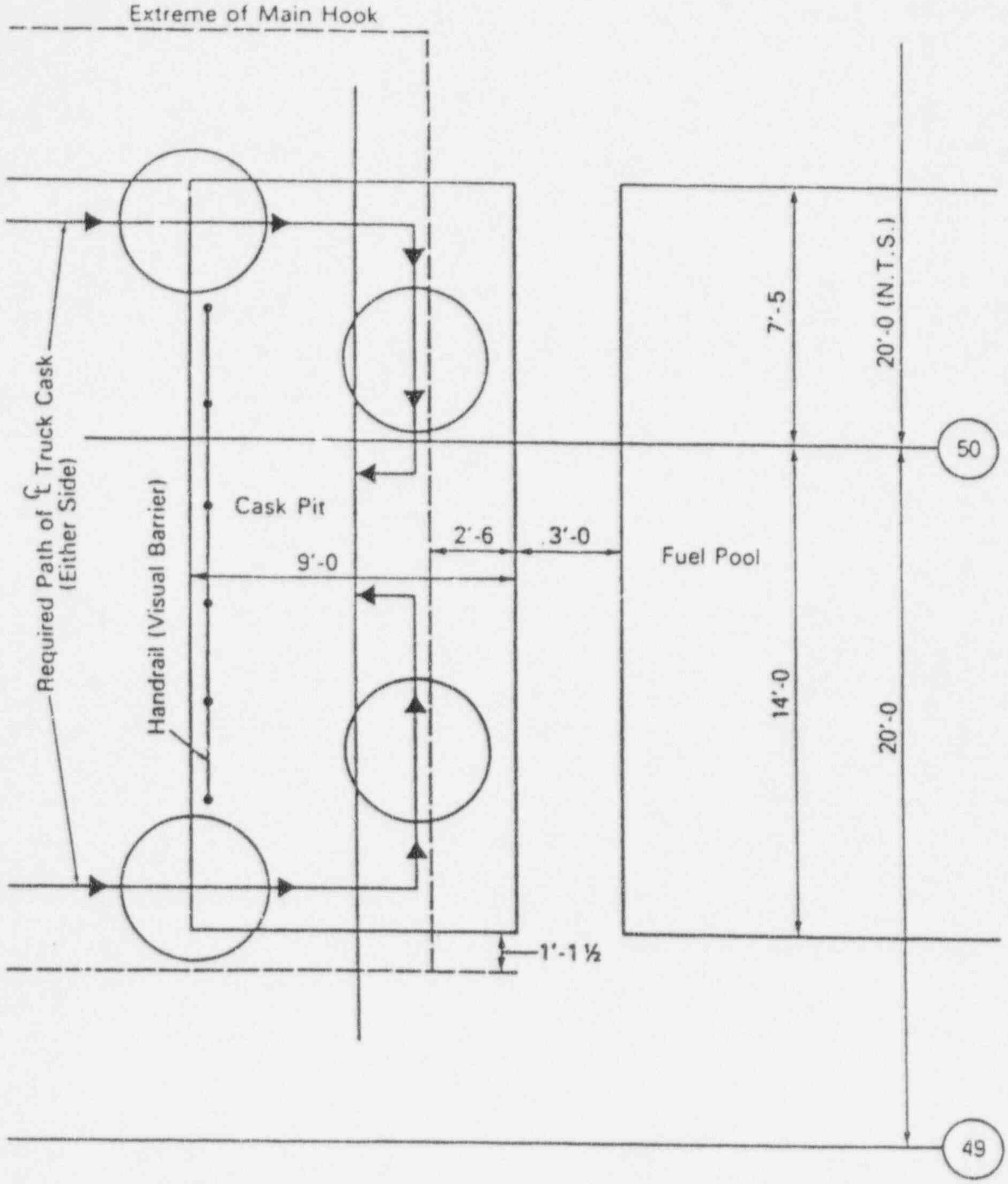

FIGURE 3.9-1
REQUIRED PATH FOR MOVEMENT OF TRUCK CASKS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or 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 once per 12 hours, one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate:

a. greater than or equal to 1000 gpm; and

b. sufficient to maintain the RCS temperature at less than or equal to 140°F.

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*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 once per 12 hours, one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate:

a. greater than or equal to 1000 gpm; and

b. sufficient to maintain the RCS temperature at less than or equal to 140°F.

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*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 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.10 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.11 FUEL HANDLING VENTILATION EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.11 The Fuel Handling Ventilation Exhaust System shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With the Fuel Handling Ventilation Exhaust System inoperable, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until the Fuel Handling Ventilation Exhaust System is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11.1 The Fuel Handling Ventilation Exhaust System shall be determined to be operating and discharging through the HEPA filters and charcoal adsorbers at least once per 12 hours whenever irradiated fuel is being moved in the storage pool and during crane operation with loads over the storage pool.

4.9.11.2 The above required Fuel Handling Ventilation Exhaust System shall be demonstrated OPERABLE:

a. 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 1% and uses the test procedure guidance of 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 35,000 cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

SURVEILLANCE REQUIREMENTS (continued)

3) Verifying a system flow rate of 35,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

b. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

c. 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 inches Water Gauge while operating the system at a flow rate of 35,000 cfm ± 10%, and

2) Verifying that the exhaust flow rate is at least 8000 cfm greater than the supply flow rate.

d. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 35,000 cfm ± 10%; and

e. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorber satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 35,000 cfm ± 10%.

3/4.9.12 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.12 The boron concentration in the spent fuel pool shall be within the limit specified in the COLR.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Immediately suspend movement of fuel assemblies in the spent fuel pool and initiate action to restore the spent fuel pool boron concentration to within its limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 Verify at least once per 7 days that the spent fuel pool boron concentration is within its limit.

3/4.9.13 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 Storage of new or irradiated fuel is limited to the configurations described in this specification.

a. New or irradiated fuel may be stored in Region 1 of the Spent Fuel Pool in accordance with these limits:

  1) Unrestricted storage of fuel meeting the criteria of Table 3.9-1; or

  2) Restricted storage in accordance with Figure 3.9-1, of fuel which does not meet the criteria of Table 3.9-1.

b. New or irradiated fuel which has decayed at least 16 days may be stored in Region 2 of the Spent Fuel Pool in accordance with these limits:

  1) Unrestricted storage of fuel meeting the criteria of Table 3.9-3; or

  2) Restricted storage in accordance with Figure 3.9-2, of fuel which meets the criteria of Table 3.9-4; or

  3) Checkerboard storage in accordance with Figure 3.9-3 of fuel which does not meet the criteria of Table 3.9-4.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Immediately initiate action to move the noncomplying fuel assembly to the correct location.

b. The provisions of Specification 3.0.3 are not applicable.

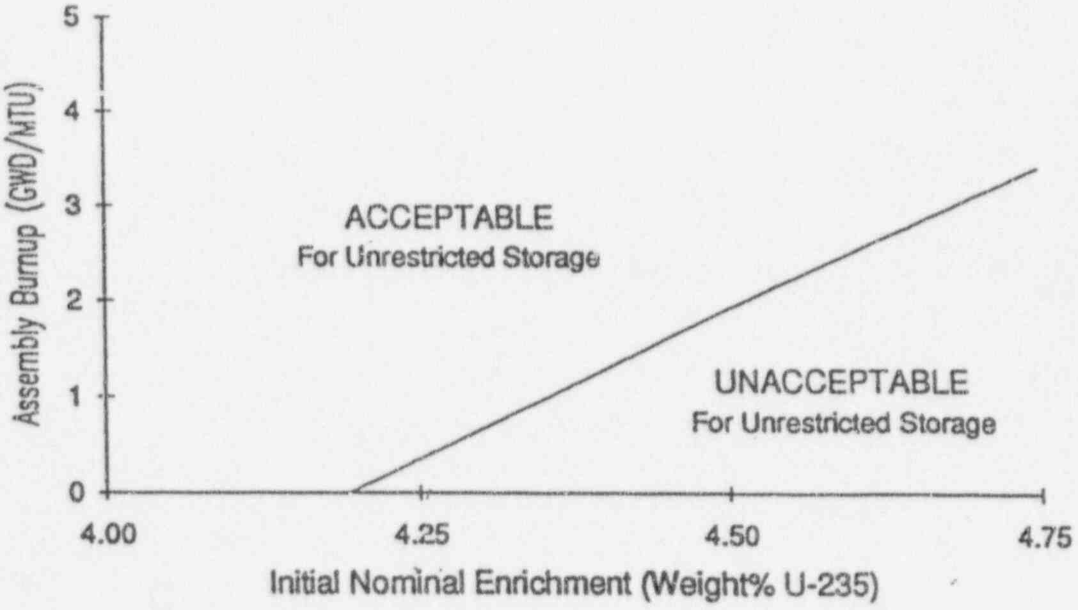
SURVEILLANCE REQUIREMENTS

4.9.13 Prior to storing a fuel assembly in the spent fuel storage pool, verify by administrative means the initial enrichment and burnup of the fuel assembly are in accordance with Specification 3.9.13.

Table 3.9-1

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Region 1 Storage

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
4.19(or less)	0
4.20	0.04
4.50	1.92
4.75	3.40



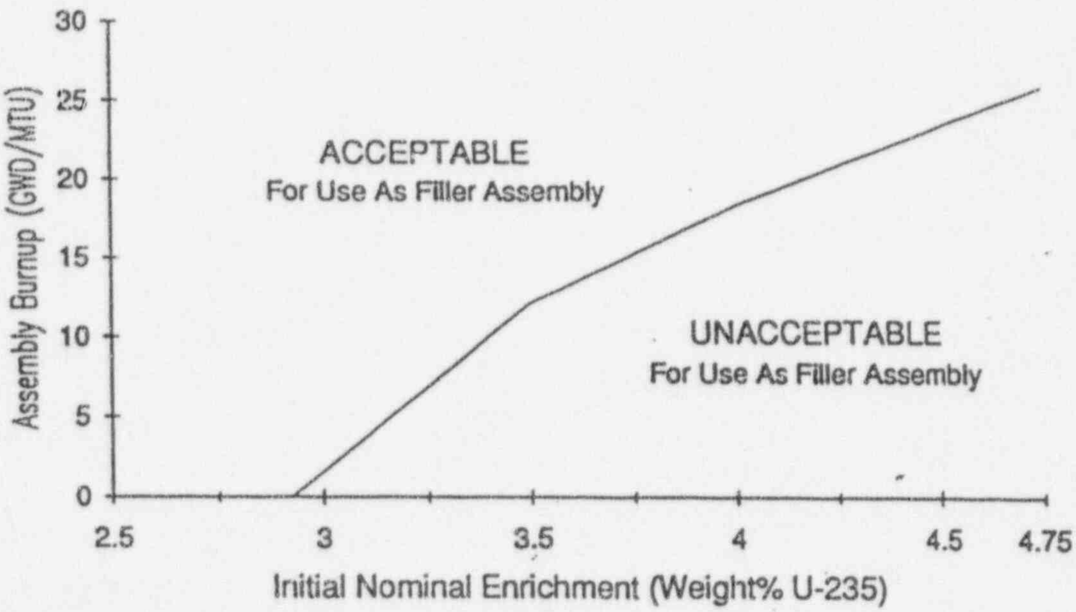
Fuel which differs from those designs used to determine the requirements of Table 3.9-1 may be qualified for Unrestricted Region 1 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Likewise, previously unanalyzed fuel up to 4.75 weight% U-235 may be qualified for Restricted Region 1 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-2

Minimum Qualifying Burnup Versus Initial Enrichment for Region 1 Filler Assemblies

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.92 (or less)	0
3.00	1.57
3.50	13.30
4.00	18.32
4.50	23.36
4.75	25.84

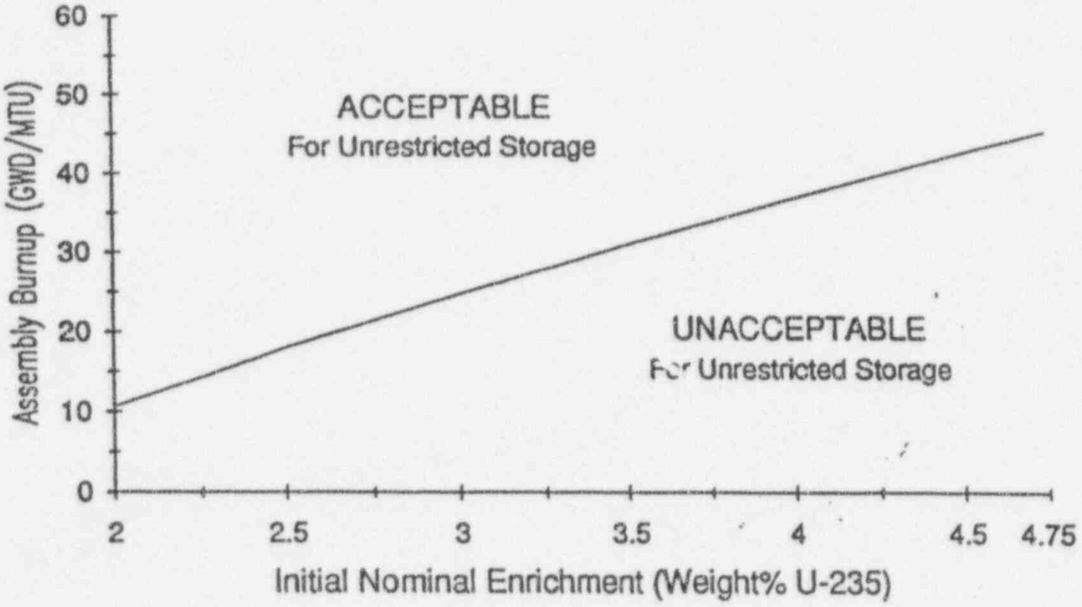


Fuel which differs from those designs used to determine the requirements of Table 3.9-2 may be qualified for use as a Region 1 Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-3

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Region 2 Storage

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00 (or less)	10.54
2.50	17.96
3.00	24.64
3.50	30.86
4.00	36.75
4.50	42.38
4.75	45.10

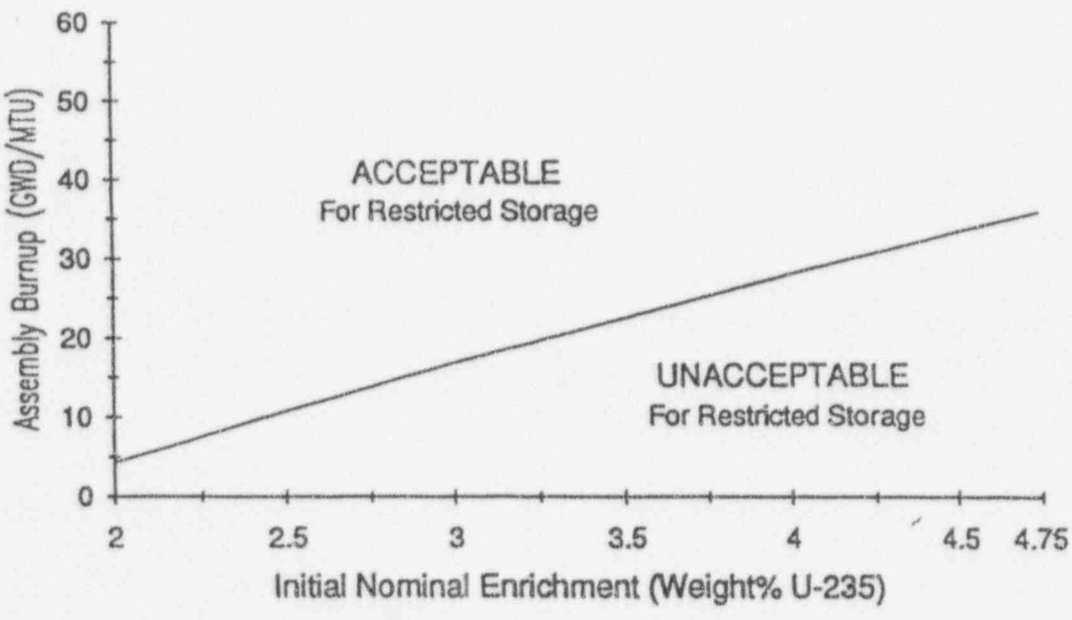


Fuel which differs from those designs used to determine the requirements of Table 3.9-3 may be qualified for Unrestricted Region 2 storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-4

Minimum Qualifying Burnup Versus Initial Enrichment for Restricted Region 2 Storage with Fillers

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00 (or less)	4.22
2.50	10.75
3.00	16.80
3.50	22.41
4.00	27.92
4.50	33.14
4.75	35.65

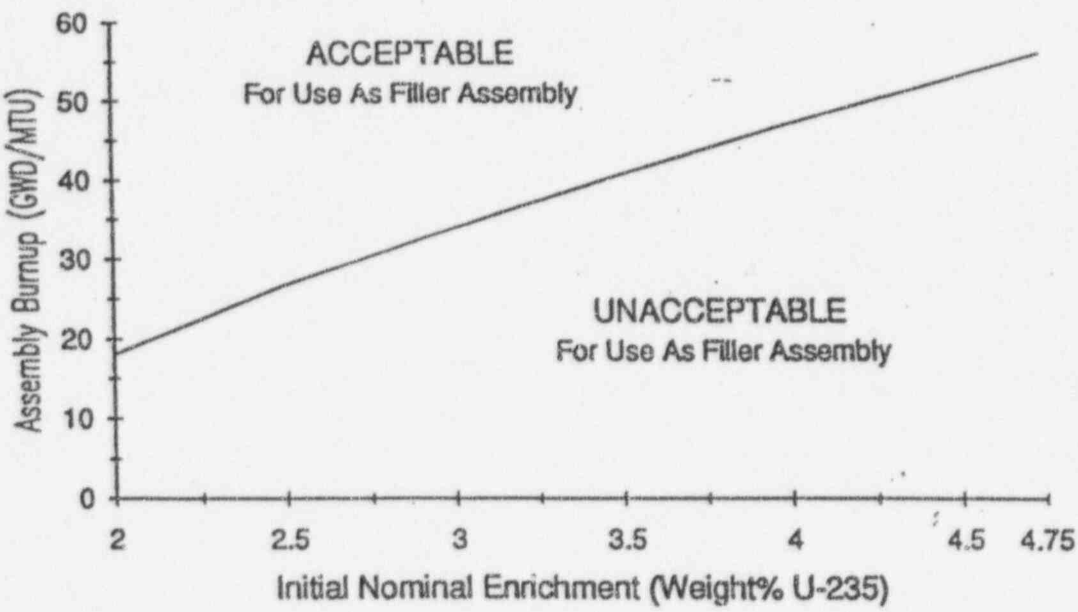


Fuel which differs from those designs used to determine the requirements of Table 3.9-4 may be qualified for Restricted Region 2 Storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Table 3.9-5

Minimum Qualifying Burnup Versus Initial Enrichment for Region 2 Filler Assemblies

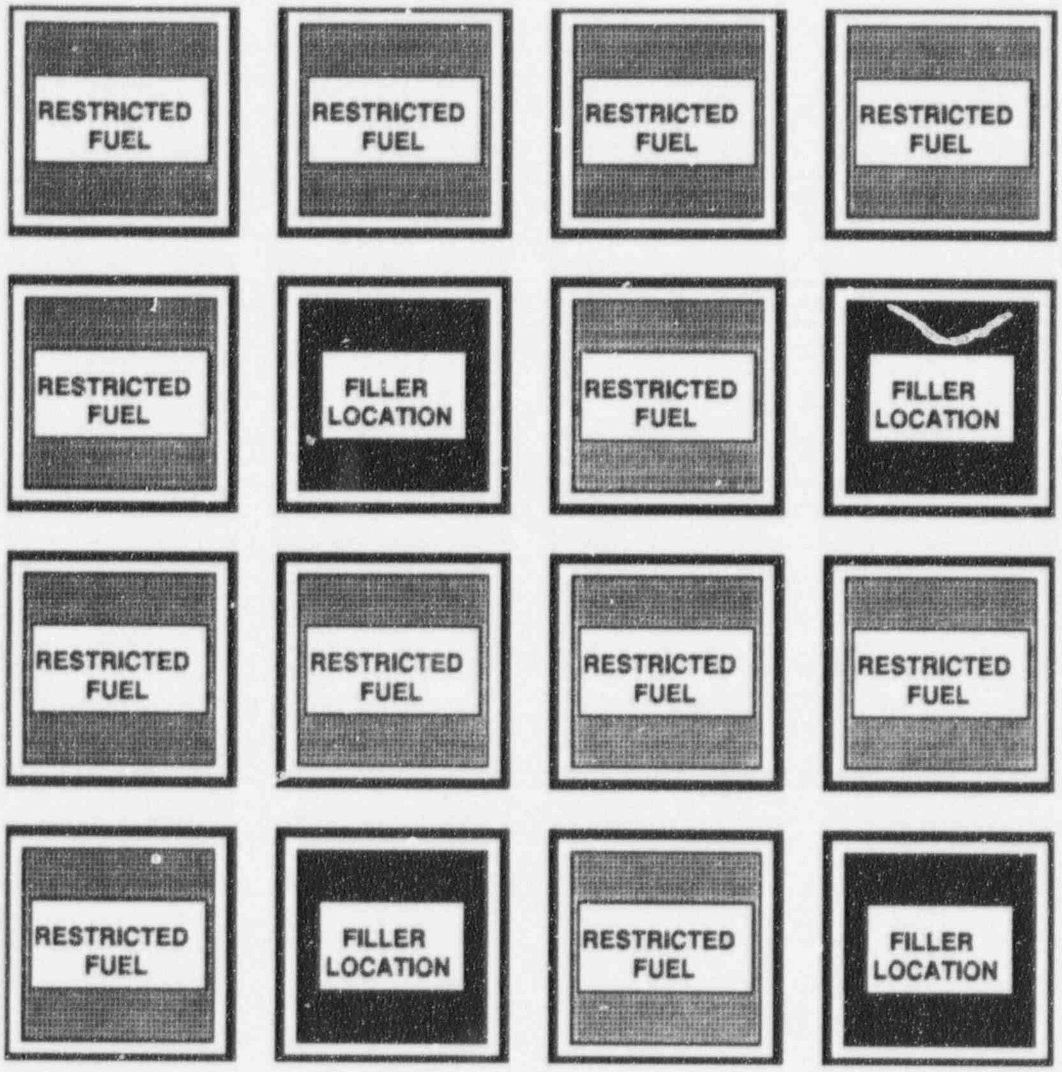
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
2.00 (or less)	18.03
2.50	26.71
3.00	33.79
3.50	40.56
4.00	46.83
4.50	52.86
4.75	55.78



Fuel which differs from those designs used to determine the requirements of Table 3.9-5 may be qualified for use as a Region 2 Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Figure 3.9-1

Required 3 out of 4 Loading Pattern
for Restricted Region 1 Storage



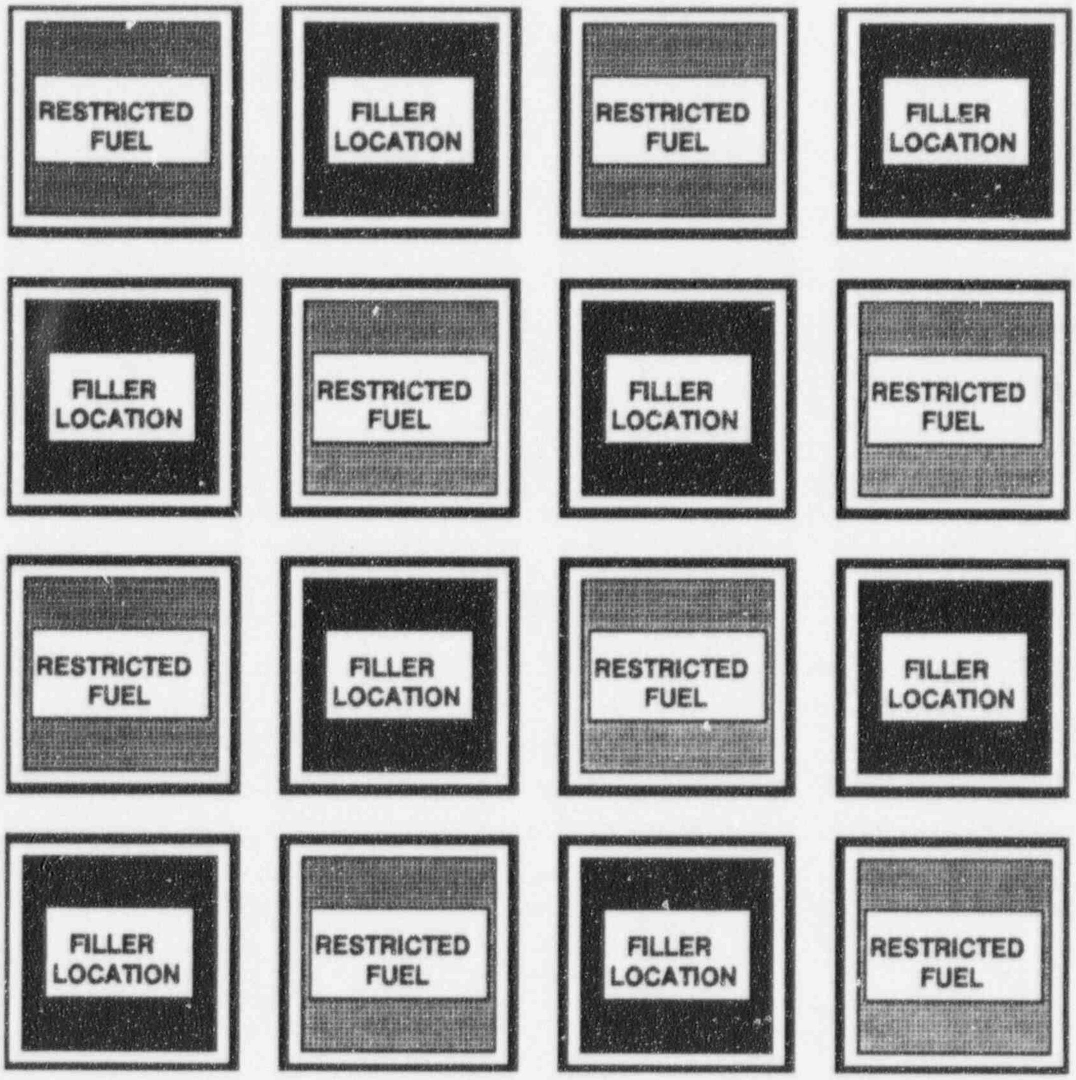
Restricted Fuel: Fuel which does not meet the minimum burnup requirements of Table 3.9-1. (Fuel which does meet the requirements of Table 3.9-1, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed)

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.9-2, or an empty cell.

Boundary Condition: Any row bounded by a Region 1 Unrestricted Storage Area shall contain a combination of restricted fuel assemblies and filler locations arranged such that no restricted fuel assemblies are adjacent to each other.
Example: In the figure above, row 1 or column 1 can not be adjacent to a Region 1 Unrestricted Storage Area, but row 4 or column 4 can be.

Figure 3.9-2

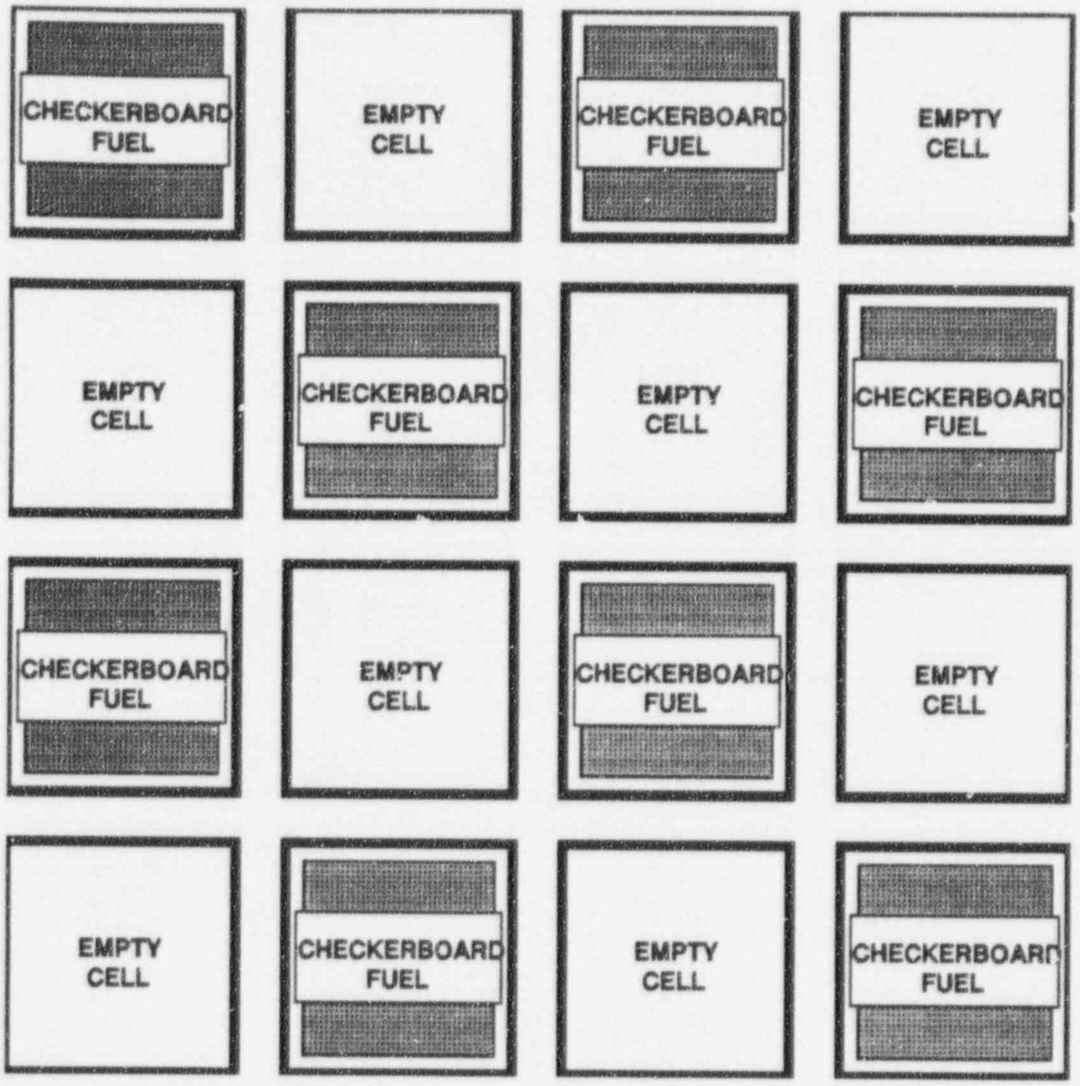
Required 2 out of 4 Loading Pattern
for Restricted Region 2 Storage



Restricted Fuel:	Fuel which meets the minimum burnup requirements of Table 3.9-4, or non-fuel components, or an empty location.
Filler Location:	Either fuel which meets the minimum burnup requirements of Table 3.9-5, or an empty cell.
Boundary Condition:	No restrictions on boundary assemblies.

Figure 3.9-3

Required 2 out of 4 Loading Pattern
for Checkerboard Region 2 Storage



Checkerboard Fuel: Fuel which does not meet the minimum burnup requirements of Table 3.9-4. (Fuel which does meet the requirements of Table 3.9-4, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed)

Boundary Condition: At least two opposite sides shall be bounded by either an empty row of cells, or a spent fuel pool wall.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Nuclear channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

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, Power Range Channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided:

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Rod Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Rod Position Indication Systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

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*This requirement is not applicable during the initial calibration of the Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 DELETED

3/4.11.1.2 DELETED

3/4.11.1.3 DELETED

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each temporary unprotected outdoor tank shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DELETED

3/4.11.2.2 DELETED

3/4.11.2.3 DELETED

3/4 11.2.4 DELETED

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, and immediately take ACTION a. above.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 49,000 Curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

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 Section 3 and 4 but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

BASES

Specifications 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of ACTION requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepared for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to

POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.4 establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

BASES

Specifications 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement 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 shall 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. These requirements apply except when relief has been provided in writing by the Commission.

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 the 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. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, 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 1.3% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections 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 End Of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value 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 EOL MTC value.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

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 pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) 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, (5) associated Heat Tracing Systems, and (6) 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 MARGIN from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions. The minimum borated water volumes and concentrations required to maintain shutdown margin for the Boric Acid Storage System and the Refueling Water Storage Tank are presented in the Core Operating Limits Report.

The Technical Specification LCO value for the Boric Acid Storage Tank and the Refueling Water Storage Tank minimum contained water volume during Modes 1-4 is based on the required volume to maintain shutdown margin, an allowance for unusable volume and additional margin as follows:

Boric Acid Storage Tank Requirements for Maintaining SDM - Modes 1-4

Required volume for maintaining SDM	presented in the COLR
Unusable volume (to maintain full suction pipe)	4,199 gallons
Additional margin	6,470 gallons

BORATION SYSTEMS (Continued)

Refueling Water Storage Tank Requirements for Maintaining SDM - Modes 1-4

Required volume for maintaining SDM	presented in the COLR
Unusable volume (below nozzle)	16,000 gallons
Additional margin	17,893 gallons

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 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Allowing two Centrifugal Charging pumps to operate simultaneously for ≤15 minutes increases the margin of safety with respect to the Reactor Coolant pump seal failure resulting in a LOCA in that the Reactor Coolant pump seal injection flow is not interrupted during pump swap. For the 15 minute period during which simultaneous Centrifugal Charging pump operation is allowed, the safety margins as related to the mass addition analysis are not appreciably reduced. Technical Specification 3.4.9.3 requires two PORVs to be operable during this period of operation, thus a mass addition transient can be relieved as required assuming the two PORVs function properly.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. The minimum borated water volumes and concentrations required to maintain shutdown margin for the Boric Acid Storage System and the Refueling Water Storage Tank are presented in the Core Operating Limits Report.

The Technical Specification LCO value for the Boric Acid Storage Tank and the Refueling Water Storage Tank minimum contained water volume during Modes 5 and 6 is based on the required volume to maintain shutdown margin, an allowance for unusable volume and additional margin as follows:

Boric Acid Storage Tank Requirements for Maintaining SDM - Modes 5 & 6

Required volume for maintaining SDM	presented in the COLR
Unusable volume (to maintain full suction pipe)	4,199 gallons
Additional margin	4,100 gallons

Refueling Water Storage Tank Requirements for Maintaining SDM - Modes 5 & 6

Required volume for maintaining SDM	presented in the COLR
Unusable volume (below nozzle)	16,000 gallons
Additional margin	6,500 gallons

BORATION SYSTEMS (Continued)

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 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The control rod insertion limit and shutdown rod insertion limits are specified in the CORE OPERATING LIMITS REPORT per specification 6.9.1.9.

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 LCO's are satisfied.

For Specification 3.1.3.1 ACTIONS c. and d., it is incumbent upon the plant personnel to verify the trippability of the inoperable control rod(s). 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.

MOVABLE CONTROL ASSEMBLIES (Continued)

During performance of the Control Rod Movement periodic test (Specification 4.1.3.1.2), there have been some "Control Malfunctions" that prohibited a control rod bank or group from moving when selected, as evidenced by the demand counters and DRPI*. In all cases, when the control malfunctions were corrected, the rods moved freely (no excessive friction or mechanical interference) and were trippable.

This surveillance test is an indirect method of verifying the control rods are not immovable or untrippable. It is highly unlikely that a complete control rod bank or bank group is immovable or untrippable. Past surveillance and operating history provide evidence of "trippability."

Based on the above information, during performance of the rod movement test, if a complete control rod bank or group fails to move when selected and can be attributed to a "Control Malfunction," the control rods can be considered "Operable" and plant operation may continue while ACTIONS c. and d. are taken.

If one or more control rods fail to move during testing (not a complete bank or group and cannot be contributed to a "Control Malfunction"), the affected control rod(s) shall be declared "Inoperable" and ACTION a. taken.

(Reference: W letter dated December 21, 1984, NS-NRC-84-2990, E. P. Rahe to Dr. C. O. Thomas)

*Digital Rod Position Indicators

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 calculated DNBR in the core at or above the design limit 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 are not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(X,Y,Z)$ Heat Flux Hot Channel Factor, is defined as the local heat flux on the surface of a fuel rod at core location X,Y,Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}(X,Y)$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along a rod at core location X,Y to the average rod power.

K(Z) is defined as the normalized $F_Q(X,Y,Z)$ limit for a given core height.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) ensure that $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ limits specified in the CORE OPERATING LIMITS REPORT (COLR) are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD envelope specified in the COLR has been adjusted for measurement uncertainty.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed ΔI power operating space during normal power operation. These alarms are active when power is greater than 50% of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, 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 ECCS acceptance criteria are not exceeded. The peaking limits are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.9.

The heat flux hot channel factor and nuclear enthalpy rise hot channel factor are each measurable, but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

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.

$F_{\Delta H}(X,Y)$ will be maintained within its limits provided Conditions a. through d. above are maintained.

The limits on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, are specified in the COLR as Maximum Allowable Radial Peaking (MARP) limits, obtained by dividing the Maximum Allowable Total Peaking (MAP) limit by the axial peak [AXIAL(X,Y)] for location (X,Y). By definition, the Maximum Allowable Radial Peaking limits will result in a DNBR for the limiting transient that is equivalent to the DNBR calculated with a design $F_{\Delta}H(X,Y)$ value of 1.50 and a limiting reference axial power shape. For transition cores, MARP limits may be applied to both MARK-BW and optimized fuel types provided allowances for differences in DNBR are accounted for in the generation of MARP limits. The MARP limits specified in the COLR include allowances for mixed core DNBR effects.

The relaxation of $F_{\Delta H}(X,Y)$ as a function of THERMAL POWER allows for a change in the radial power shape for all permissible control bank insertion limits. This relaxation is implemented by the application of the following factors:

$k = [1 + (1/RRH)\ (1 - P)]$

where k = power factor multiplier applied to the MAP limits
p = THERMAL POWER/RATED THERMAL POWER
RRH is given in the COLR

BASES

HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The hot channel factor $F^M_Q(X,Y,Z)$, and the nuclear enthalpy rise hot channel factor, $F^M_{\Delta H}(X,Y)$, are measured periodically to verify that the core is operating as designed. $F^M_Q(X,Y,Z)$ and $F^M_{\Delta H}(X,Y)$ are compared to allowable limits to provide reasonable assurance that limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety Systems Settings), 3.1.3 (Movable Control Assemblies), 3.2.1 (Axial Flux Difference), and 3.2.4 (Quadrant Power Tilt Ratio). A peaking margin calculation is performed to provide the basis for decreasing the width of the AFD and $f(\Delta I)$ limits and for reducing THERMAL POWER.

When an $F^M_Q(X,Y,Z)$ measurement is obtained from a full-core map in accordance with surveillance requirements of Specification 4.2.2, no uncertainties are applied to the measured peak since a measurement uncertainty of 5.0% and a manufacturing tolerance of 3.0% are included in the peaking limit. When $F^M_Q(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2, the measured peak is increased by the radial-local peaking factor and appropriate allowances for measurement uncertainty and for manufacturing tolerances.

When an $F^M_{\Delta H}(X,Y)$ measurement is obtained from a full-core map, regardless of the reason, no uncertainties are applied to the measured peak since the required uncertainties are included in the peaking limit.

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 the x-y plane power tilts. The peaking increase that corresponds to a QUADRANT POWER TILT RATIO of 1.02 is included in the generation of the AFD limits.

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 rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q(X,Y,Z)$ is reinstated by reducing the power by 3% from RATED THERMAL POWER for each percent of tilt in excess of 2.0%.

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.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design limit DNBR throughout each analyzed transient. As noted on Figure 3.2-1, RCS flow rate and THERMAL POWER may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the power level is decreased) to ensure that the calculated DNBR will not be below the design DNBR value. The relationship defined on Figure 3.2-1 remains valid as long as the limits placed on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, in Specification 3.2.3 are maintained. The indicated T_{avg} values and the indicated pressurizer pressure values correspond to analytical limits of 592.6°F and 2220 psia respectively, with allowance for indication instrumentation measurement uncertainty. When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-1 since an RCS total flow rate measurement uncertainty, greater than or equal to the value stated on Figure 3.2-1 has been allowed for in determination of the design DNBR value.

3/4.2.5 DNB PARAMETERS (Continued)

The measurement error for RCS total flow rate is based upon the performance of past precision heat balances. Sets of elbow tap coefficients, as determined during these heat balances, were averaged for each elbow tap to provide a single set of elbow tap coefficients for use in calculating RCS flow. This set of coefficients establishes the calibration of the RCS flow rate indicators and becomes the set of elbow tap coefficients used for RCS flow measurement. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from these heat balances in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-1. 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 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. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip and Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features Instrumentation and (3) sufficient system functions 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 accident analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. The NRC Safety Evaluation Reports for the WCAP-10271 series were provided in letters dated February 21, 1985 from C. O Thomas (NRC) to J. J. Sheppard (WOG), February 22, 1989 from C. E. Rossi (NRC) to R. A. Newton (WOG), and April 30, 1990 from C. E. Rossi (NRC) to G. T. Goering (WOG).

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Feature actuation associated with each channel is completed within the time limit assumed in the accident 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 times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, and (10) nuclear service water pumps start and automatic valves position.

Technical Specifications for the Reactor Trip Breakers and the Reactor Trip Bypass Breakers are based upon NRC Generic Letter 85-09 "Technical Specifications for Generic Letter 83-28, Item 4.3," dated May 23, 1985.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 Defeats the manual block of Safety Injection actuation on low pressurizer pressure and low steamline pressure and defeats steamline isolation on negative steamline pressure rate. Defeats the manual block of the motor-driven auxiliary feedwater pumps on trip of main feedwater pumps and low-low steam generator water level.

P-12 On 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 level, P-14 automatically trips all feedwater isolation valves and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 Deleted

3/4.3.3.4 Deleted

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY 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 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations "

3/4.3.3.7 DELETED

3/4.3.3.8 NOT USED

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The gas instrumentation is provided for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM.

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat; however, single failure considerations require that three loops be OPERABLE. Also, the uncontrolled bank withdrawal from zero power or subcritical assumes three reactor coolant loops in operation.

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 a reactor coolant pump with one or more RCS cold legs less than or equal to 300°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.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the 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 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no Reactor trip until the first Reactor Trip System 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. Specifications 3.4.2.1 and 3.4.2.2 allow a + 3% and - 2% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during surveillance testing to allow for drift.

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. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions: 1) Manual control of PORVs to control RCS pressure. This is a function that is used for the steam generator tube rupture accident coincident with a loss of all offsite power and for plant shutdown. 2) Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified

3/4.4.4 RELIEF VALVES (Continued)

reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of RCS pressure and isolate a PORV with excessive leakage. 4) Automatic control of PORVs to control RCS pressure. This is a function that reduces challenges to the code safety valves for overpressurization events. 5) Manual control of a block valve to isolate a stuck-open PORV.

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 B&W process (or method) equivalent to the inspection method described in Topical Report BAW-2045(P)-A will be used. Inservice inspection of steam generator sleeves is also required to ensure RCS integrity. Because the sleeves introduce changes in the wall thickness and diameter, they reduce the sensitivity of eddy current testing, therefore, special inspection methods must be used. A method is described in Topical Report BAW-2045(P)-A with supporting validation data that demonstrates the inspectability of the sleeve and underlying tube. As required by NRC for licensees authorized to use this repair process, McGuire commits to validate the adequacy of any system that is used for periodic inservice inspections of the sleeves, and will evaluate and, as deemed appropriate by Duke Power Company, implement testing methods as better methods are developed and validated for commercial use.

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

STEAM GENERATORS (Continued)

will be found during scheduled inservice steam generator tube examinations. Repair will be required for all tubes with imperfections exceeding the repair limit of 40% of the tube nominal wall thickness. Installed sleeves with imperfections exceeding 40% of the sleeve nominal wall thickness will be plugged. Defective steam generator tubes can be repaired by the installation of sleeves which span the area of degradation, and serve as a replacement pressure boundary for the degraded portion of the tube, allowing the tube to remain in service. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness. For tubes with degradation below the F* distance, and not degraded within the F* distance, repair is not required. If a tube is sleeved due to degradation in the F* distance, then any defects in the tube below the sleeve will remain in service without repair.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73 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

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 Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

OPERATIONAL LEAKAGE (continued)

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 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the accident analyses.

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.

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.

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 Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm. The values for the limits on specific activity represent limits based upon a parametric

SPECIFIC ACTIVITY (Continued)

evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the McGuire 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 1.0 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.

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 micro-Curie/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 radio-iodine contribution would probably be about 20%. The exclusion of radio-nuclides 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 radionu-clides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective ACTION. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of the effective full power years (EFPY) of service life identified on the applicable technical specification figure. The 16 EFPY service life period continues to ensure that the limiting RT_{NDT} at the 1/4T location in the core region is a bounding value. 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 phosphate content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT}. The adjusted reference temperature has been computed by Regulatory Guide 1.99, Rev. 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 the identified service life. Adjustments for possible errors in the pressure and temperature sensing instruments are included when stated on the applicable figure.

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 50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the pressure vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the pressure vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS

COMPONENT	CU (%)	P (%)	NDTT (°F)	50 FT-LB/35 MIL TEMP (°F)		RT_{NDT} (°F)	MIN. UPPER SHELF (FT-LB)	
				PARALLEL TO MAJOR WORKING DIRECTION	NORMAL TO MAJOR WORKING DIRECTION(b)		PARALLEL TO MAJOR WORKING DIRECTION	NORMAL TO MAJOR WORKING DIRECTION
Cl.Hd. Dome	-	-	+20	77	138*	78	100(b)	-
Cl. Hd. Seg.	-	-	+10	27	76*	16	135	88**
Cl. Hd. Flange	-	-	+48$	-50	-5*	48	154	100**
Vessel Flange	-	-	+40$	-42	-13*	40	153	99**
Inlet Nozzle	-	-	+60$	+22	44*	60	129	84**
Inlet Nozzle	-	-	+60$	41	56*	60	132	86**
Inlet Nozzle	-	-	+60$	30	61*	60	121	79**
Inlet Nozzle	-	-	+60$	40	69*	60	115	75**
Outlet Nozzle	-	-	+60$	-15	24*	60	124	81**
Outlet Nozzle	-	-	+60$	18	52*	60	114	74**
Outlet Nozzle	-	-	+60$	4	46*	60	133	86**
Outlet Nozzle	-	-	+60$	72	101*	60	122	79**
Upper Shell	0.14	-	+10	55	108*	48	109	71**
Upper Shell	0.10	-	+10	67	106*	46	102(c)	J66(c)
Upper Shell	0.13	-	0	40	76*	16	141	92**
Inter. Shell	0.13	0.010	-30	36	94	34	136	98
Inter. Shell	0.14	0.011	0	33	60	0	137	102
Inter. Shell	0.11	0.013	-20	12	47	-13	153	103
Lower Shell	0.14	0.009	-10	36	60	0	125	93
Lower Shell	0.10	0.010	-10	32	90	30	136	113
Lower Shell	0.10	0.010	0	42	75	15	128	100

TABLE B 3/4.4-1 (Continued)
REACTOR VESSEL TOUGHNESS

COMPONENT	CU (%)	P (%)	NDTT (°F)	50 FT-LB/35 MIL TEMP (°F) PARALLEL TO MAJOR WORKING DIRECTION	50 FT-LB/35 MIL TEMP (°F) NORMAL TO MAJOR WORKING DIRECTION(b)	RT_{NDT} (°F)	MIN. UPPER SHELF (FT-LB) PARALLEL TO MAJOR WORKING DIRECTION	MIN. UPPER SHELF (FT-LB) NORMAL TO MAJOR WORKING DIRECTION
Bot. Peel Seg.	-	-	-70	14	45*	-15	137	89**
Bot. Peel Seg.	-	-	-30	25	58*	-2	145	94**
Bot.Peel Seg.	-	-	-20	42	87*	27	123	80**
Bot. Hd. Dome	-	-	0	50	101*	41	123	80**
Weld	0.30(a)	-	-60	-	-6	-50	-	110
Haz	-	-	-50	-	28	-32	-	80

*Estimated (77 ft-lb/54 mil temp. for longitudinal data)
**Estimated (65% if longitudinal upper shelf)
$Estimated (60°F or 100 ft-lb temp., whichever is less)

(a) Conservative Estimate
(b) 95% Shear
(c) 90% Shear

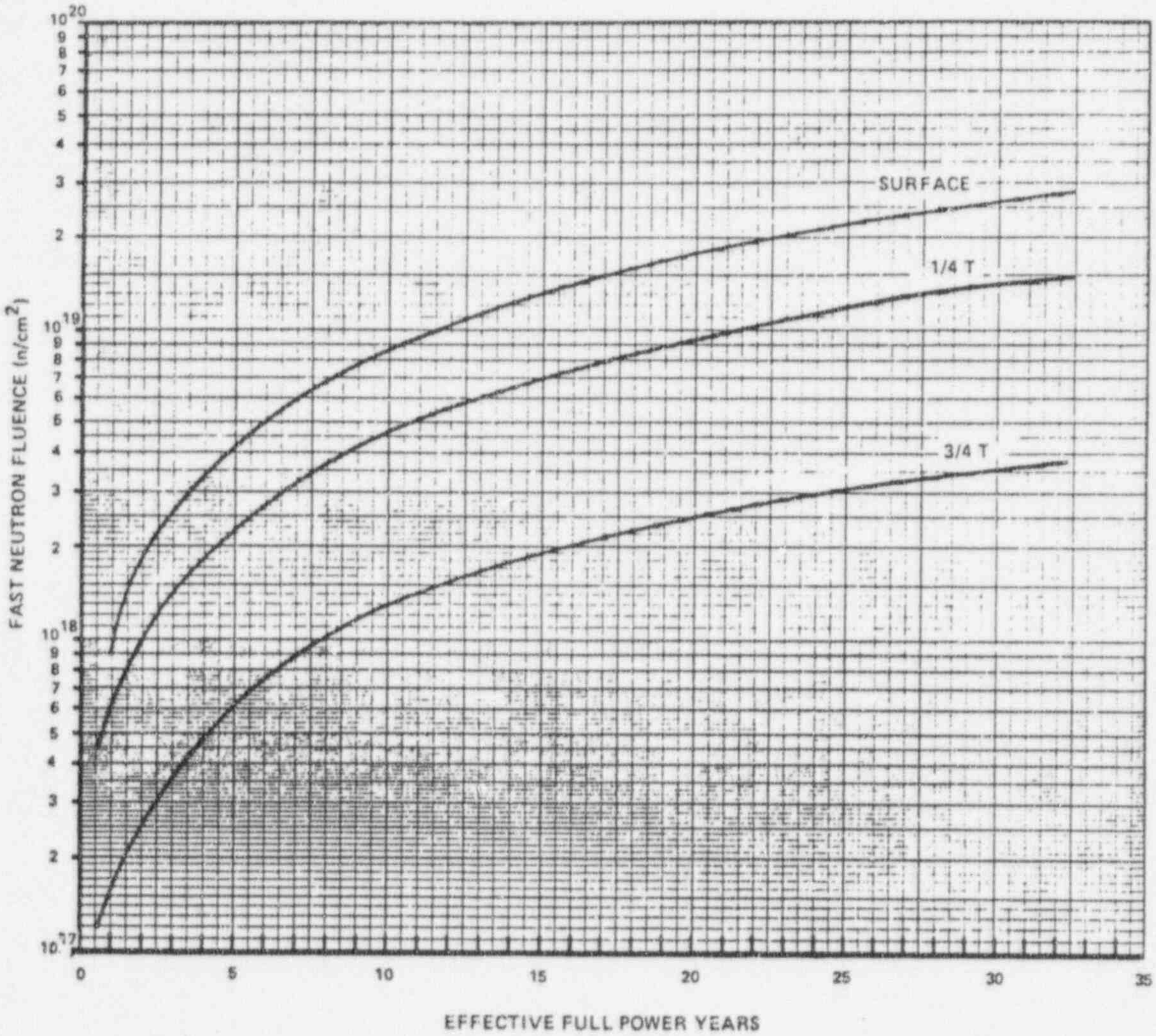

FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E>1MEV)
AS A FUNCTION OF EFFECTIVE FULL POWER YEARS

PRESSURE/TEMPERATURE LIMITS (Continued)

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical 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 non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{IT} \leq K_{IR} \qquad (2)$$

Where: K_{IM} is the stress intensity factor caused by membrane (pressure) stress,

K_{IT} is the stress intensity factor caused by the thermal gradients,

K_{IR} is 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

PRESSURE/TEMPERATURE LIMITS (Continued)

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 delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T 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

PRESSURE/TEMPERATURE LIMITS (Continued)

end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves in technical specifications for the heatup rate data and the cooldown rate data may be adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves. Where technical specification curves have not been adjusted, such adjustments are made by plant procedures.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.75 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 300°F. Either of

PRESSURE/TEMPERATURE LIMITS (Continued)

the PORVs or the RCS vent opening 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 HPSI pump and its injection into a water-solid RCS. The Pressurizer PORV setpoints for low temperature overpressure protection are based on limiting the peak pressure during the limiting transient to 1.10 times the ASME Section XI, Appendix G limits, in accordance with ASME code case N-514.

Credit is taken for the RHR suction relief valve (ND-3) during conditions where relieving capacity at rated accumulation is sufficient to prevent exceeding the above allowable pressure limits.

Cooldown limits/minimum RCS temperature restrictions ensure the allowable pressure limits will not be exceeded.

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 Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 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, 1971 Edition and Addenda through Winter 1972.

3/4.4.11 REACTOR VESSEL HEAD VENT SYSTEM

Reactor Vessel Head Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function. (Operability of the pressurizer steam space vent path is provided by Specifications 3/4.4.4 and 3/4.4.9.3.)

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 surveillance to verify Reactor Vessel Head Vent flowpath is qualitative as no specific size or flow rate is required to exhaust noncondensible gases. The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) Cold Leg 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 allowed outage time for the accumulators are variable based upon boron concentration to ensure that the reactor is shutdown following a LOCA and that any problems are corrected in a timely manner. The minimum boron concentration required to ensure post-LOCA subcriticality as presented in the Core Operating Limits Report, is based on nominal accumulator volume conditions and allows additional outage time since subcriticality is assured when the boron concentration is above this value. A slightly high boron concentration, the minimum accumulator boron concentration limit for LCO 3.5.1c presented in the Core Operating Limits Report, is based on worst case liquid mass, boron concentration and measurement errors. A concentration less than this LCO value in any single accumulator or as a volume weighted average may be indicative of a problem, such as valve leakage. Since reactor shutdown is assured if the boron concentration is above the minimum concentration to ensure post-LOCA subcriticality and the accumulator volume is greater than or equal to the nominal volume, additional time is allowed to restore boron concentration in the accumulators.

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.

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 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Allowing two Centrifugal Charging pumps to operate simultaneously for ≤15 minutes increases the margin of safety with respect to the Reactor Coolant pump seal failure resulting in a LOCA in that the Reactor Coolant pump seal injection flow is not interrupted during pump swap. For the 15 minute period during which simultaneous Centrifugal Charging pump operation is allowed, the safety margins as related to the mass addition analysis are not appreciably reduced. Technical Specification 3.4.9.3 requires two PORVs to be operable during this period of operation, thus a mass addition transient can be relieved as required assuming the two PORVs function properly.

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 [Deleted]

3/4.5.5 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 7.5 and 9.5 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.

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 accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 1.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 15 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 14.5 psig. The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 14.8 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that: (1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions, and (2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 100°F for the lower compartment, 75°F for the upper compartment and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to 14.8 psig which is less than the containment design pressure of 15 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 15 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 REACTOR BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment reactor building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide: (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

3/4.6.1.8 ANNULUS VENTILATION SYSTEM

The OPERABILITY of the Annulus Ventilation System ensures that during LOCA conditions, containment vessel leakage into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The specified laboratory test method, ASTM-3803-89, implies that heaters may be unavailable for controlling the relative humidity of the influent air entering the charcoal absorber section to ≤ 70 percent. This is acceptable since the accident analysis with appropriate absorber efficiencies for radioiodine in elemental and organic forms based on the above test shows that the control room radiation doses to be within the 10 CFR Part 50, Appendix A, GDC 19 limits during design basis LOCA conditions. However, specifications are included to ensure heater operability and corrections ACTIONS are identified to address the contingency of inoperable heaters; these are in place to increase the safety margin of the filters. Heater operation is not necessary to meet the assumptions used in the accident analyses and limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during LOCA conditions.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves for the lower compartment (24-inch) and instrument room (12-inch and 24-inch) are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the purge supply and exhaust isolation valves in the upper compartment (24-inch) since, unlike the valves in the lower compartment and instrument room, the upper compartment valves will close during a LOCA. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation with these valves open will be limited to 250 hours during a calendar year.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM (Continued)

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

CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

Containment isolation valves are listed in FSAR Table 6.2.4-1. Those valves with a required isolation time have a value given in the "MAX ISOLATION TIME (SEC)" column. Penetration test type (type B, type C, or None) is listed in the "TEST TYPE" column of the table for each containment penetration. Changes to the FSAR are made in accordance with 10 CFR 50.59.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

3/4.6.4 COMBUSTIBLE GAS CONTROL (Continued)

The OPERABILITY of at least 64 of 66 igniters ensures that the Distributed Ignition System will maintain an effective coverage throughout the containment. This system of igniters will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 14.8 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will: (1) be distributed evenly through the containment bays, (2) contain sufficient boron to preclude dilution of the containment sump following the LOCA, and (3) contain sufficient heat removal capability to condense the Reactor Coolant System volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1081 pounds of ice per basket contains a 10% conservative allowance for ice loss through sublimation which is a factor of 10 higher than assumed for the ice condenser design. The minimum weight figure of 2,099,790 pounds of ice also contains an additional 1.1% conservative allowance to account for systematic error in weighing instruments. In the event that observed sublimation rates are equal to or lower than design predictions after 3 years of operation, the minimum ice baskets weight may be adjusted downward. In addition, the number of ice baskets required to be weighed each 9 months may be reduced after 3 years of operation if such a reduction is supported by observed sublimation data.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the Ice Bed Temperature Monitoring System ensures that the capability is available for monitoring the ice temperature. In the event the system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.3 ICE CONDENSER DOORS

The OPERABILITY of the ice condenser doors and the requirement that they be maintained closed ensures that the Reactor Coolant System fluid released during a LOCA will be diverted through the ice condenser bays for heat removal and that excessive sublimation of the ice bed will not occur because of warm air intrusion.

If an ice condenser door is not capable of opening automatically, then system function is seriously degraded and immediate action must be taken to restore the opening capability of the door. Not capable of opening automatically is defined as those conditions in which a door is physically blocked from opening by installation of a blocking device or by obstruction from temporary or permanent installed equipment or is otherwise inhibited from opening such as may result from ice, frost, debris or increased door opening torque.

3/4.6.5.4 INLET DOOR POSITION MONITORING SYSTEM

The OPERABILITY of the Inlet Door Position Monitoring System ensures that the capability is available for monitoring the individual inlet door position. In the event the system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.5 DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

The requirements for the divider barrier personnel access doors and equipment hatches being closed and OPERABLE ensure that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of the steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.6.5.6 CONTAINMENT AIR RETURN AND HYDROGEN SKIMMER SYSTEM

The OPERABILITY of the Containment Air Return and Hydrogen Skimmer System ensures that following a LOCA: (1) the containment atmosphere is circulated for cooling by the Spray System, and (2) the accumulation of hydrogen in localized portions of the containment structure is minimized.

3/4.6.5.7 and 3/4.6.5.8 FLOOR AND REFUELING CANAL DRAINS

The OPERABILITY of the ice condenser floor and refueling canal drains ensures that following a LOCA, the water from the melted ice and Containment Spray System has access for drainage back to the containment lower compartment and subsequently to the sump. This condition ensures the availability of the water for long-term cooling of the reactor during the post-accident phase.

3/4.6.5.9 DIVIDER BARRIER SEAL

The requirement for the divider barrier seal to be OPERABLE ensures that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.7 PLANT SYSTEMS

BASES

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 Coolant System pressure will be limited to within 110% 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. Table 3.7-3 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during surveillance testing to allow for drift. The total relieving capacity for all valves on all of the steam lines is 15.9 x 10^6 lbs/hr which is 105% of the total secondary steam flow of 15.14 x 10^6 lbs/hr 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 Tables 3.7-1 and 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:

$$SP = \frac{(X) - (Y)(U)}{X} \times (*)$$

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,

* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for three loop operation. This value left blank pending NRC approval of 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 conditions in the event of a total loss-of-offsite power.

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1210 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 900 gpm at a pressure of 1210 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

Verification of the steam turbine-driven pump discharge pressure should be deferred until suitable test conditions are established (i.e., greater than or equal to 900 psig in the secondary side of the steam generator). This deferral is required because until 900 psig is reached, there is insufficient steam pressure to perform the test.

3/4.7.1.3 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.4 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the accident 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 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 NUCLEAR SERVICE WATER SYSTEM

The OPERABILITY of the Nuclear Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits. Periodic flow balance tests, delta-P tests, and heat balance tests are performed as required to assure adequate flow to all essential heat exchangers for which flow instrumentation is provided.

Portions of the Nuclear Service Water System are common to both units. These shared portions of the system are indicated on Figure 3/4 7-1 and common valves are listed in Table B 3/4 7-1 and include common suction piping and cross-connect piping as indicated on the figure.

3/4.7.4 NUCLEAR SERVICE WATER SYSTEM (Continued)

With the exception of ORN-1 all shared valves receive emergency power from two essential motor control centers (1EMXH, 2EMXH). ORN-1 is normally open with the power removed. Motor Control Center (MCC) 1EMXH can be powered by either Unit 1 or Unit 2 A Train Emergency D/G's via their associated switchgear. Motor Control Center 2EMXH can be powered by either Unit 1 or 2 B Train Emergency D/G's via their associated switchgear.

The four loops (two per unit) ensure redundancy and the availability of cooling to both units, even if a single failure were to render two loops inoperable. (Such a failure would involve train A of both units or train B of both units, not both trains of the same unit). The Action statements are separated to clarify that portions of the systems are shared though the majority of each of the four loops is independent.

In the event of a safety injection or blackout signal on either unit, train A of both units will align to Lake Norman and train B of both units will align to the SNSWP. Additionally, all train A to train B cross-connects will close on both units as will non-safety to safety related cross-connects. These actuations assure independence of the loops and the required redundancy under design basis conditions.

3/4.7.5 STANDBY NUCLEAR SERVICE WATER POND

The limitations on the standby nuclear service water pond level and temperature ensure that sufficient cooling capacity is available either to: (1) provide normal cooldown of the facility, or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974. The Surveillance Requirements specified for the dam inspection will conform to the recommendations of Regulatory Guide 1.127, Revision 1, March 1978

TABLE B 3/4 7-1

UNITS 1 AND 2
NUCLEAR SERVICE WATER SYSTEM SHARED VALVES

ORN-1	LOW LEVEL INTAKE SUP TO RN
ORN-2B	TRAIN A RC SUPPLY
ORN-3A,C	TRAIN A RC SUPPLY
ORN-4A,C	TRAIN B RC SUPPLY
ORN-5B	TRAIN B RC SUPPLY
ORN-7A,C	TRAIN A SNSWP SUPPLY
ORN-9B	TRAIN B SNSWP SUPPLY
ORN-10A,C	TRAIN B LLI SUPPLY
ORN-11B	TRAIN B LLI SUPPLY
ORN-12A,C	TRAIN A LLI SUPPLY
ORN-13A	TRAIN A LLI SUPPLY
ORN-14A	TRAIN A SUCT X-CONNECT
ORN-15B	TRAIN B SUCT X-CONNECT
ORN-147A,C	TRAIN A DISCH TO RC
ORN-148A,C	TRAIN A DISCH TO RC
ORN-149A	TRAIN A DISCH TO RC
ORN-150A,C	TRAIN A DISCH X-CONNECT
ORN-151B	TRAIN B DISCH X-CONNECT
ORN-152B	TRAIN B DISCH TO SNSWP
ORN-283A,C	TRAIN B DISCH TO RC
ORN-284B	TRAIN B DISCH TO RC
ORN-301A,C	RV SUPPLY FROM LLI
ORN-302B	RV SUPPLY FROM LLI

3/4.7.6 CONTROL AREA VENTILATION SYSTEM

The OPERABILITY of the Control Area Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The specified laboratory test method, namely, ASTM D3803-89, implies that heaters may be unavailable for controlling the relative humidity of the influent air entering the charcoal absorber section to ≤ 70 percent. This is acceptable since accident analysis with appropriate absorber efficiencies for radioiodine in elemental and organic forms based on the above test shows the site boundary radiation doses to be within the 10 CFR Part 100 limits during design basis LOCA conditions. However, specifications are included to ensure heater operability and corrective ACTIONS are identified to address the contingency of inoperable heaters; these are in place to increase the safety margin of the 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 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR 50. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.7 AUXILIARY BUILDING FILTERED VENTILATION EXHAUST SYSTEM

The OPERABILITY of the Auxiliary Building Filtered Ventilation Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the auxiliary building following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations were assumed in the accident analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. The methyl iodide penetration test criterion for the carbon samples has been established at 10% (i.e., 90% removal) which is greater than the iodine removal in the accident analysis.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program 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.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2 kip, 10 kip, and 100 kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured Company "B" for the purposes of this specification would be of a different type, as would hydraulic snubbers from either manufacturer.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide assurance of snubber functional reliability one of the three sampling and acceptance criteria methods are used:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or continue testing* 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 snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date.

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*If testing continues to between 100-200 snubbers (or 1-2 weeks) and still the accept region has not been reached, then the actual % of population quality (C/N) should be used to prepare for extended or 100% testing.

SNUBBERS (Continued)

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, 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. The requirements for the maintenance of records and the snubber service life review not intended to affect plant operation.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 Deleted

3/4.7.11 Deleted

3/4.7.12 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 3.9°F.

3/4.7.13 GROUNDWATER LEVEL

This Technical Specification is provided to ensure that groundwater levels will be monitored and prevented from rising to the potential failure limit for the McGuire Units 1 and 2 Auxiliary Buildings. This potential failure limit is based on engineering calculations that have determined that the Auxiliary Buildings are susceptible to overturning due to buoyancy at elevation 737 feet Mean Sea Level (MSL). Under the requirements of this Technical Specification, if groundwater level exceeds elevation 731 feet MSL, (3 out of 5 Tech Spec

GROUNDWATER LEVEL (Continued)

groundwater monitor alarms), and cannot be reduced in one (1) hour, McGuire must begin reducing Units 1 and 2 to Mode 5, Cold Shutdown.

Analysis performed by Design Engineering determined that the Reactor and Diesel Generator Buildings are designed to withstand hydrostatic loadings due to groundwater levels up to elevation 760 feet MSL; therefore, no Technical Specification requirements are specified for these structures.

Elevation 731 feet MSL is the Technical Specification action level of the five Technical Specification groundwater monitors listed in Table 3.7-7. The East Wall exterior monitor alarm at elevation 731 feet MSL is the Alert alarm. The other four (4) monitors are Hi-Hi alarms at elevation 731 feet (MSL).

The East Wall exterior monitor was originally on the exterior of the Unit 2 Auxiliary Building and subsequently was enclosed by the construction of the Equipment Staging Building.

As required by Operations procedures, any alarms on non-Technical Specification groundwater monitors will also be investigated. Additionally, if three (3) out of the five (5) Technical Specification groundwater monitors alarm at levels below the Technical Specification action levels, Operations will contact Duke Engineering (Civil) for investigation and resolution of the increased groundwater level.

If one or more of the 5 Technical Specification groundwater monitors is determined to be inoperable, the monitor(s) will be considered to be indicating above the 731'-0" level until repaired and returned to an operable status.

3/4.8.1, 3/4.8.2 AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

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 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 ACTION requirements for diesel generator testing in the event of the inoperability of other electric power sources also reflect the potential for degradation of the diesel generator due to excessive testing. This concern has developed, concurrently with increased industry experience with diesel generators, and has been acknowledged by the NRC staff in Generic Letter 84-15.

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.

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

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; also, Generic Letter 84-15, which modified the testing frequencies specified in Regulatory Guide 1.108.

Some of the Surveillance Requirements for demonstrating the operability of the diesel generators are modified by a footnote. The Specifications state the Surveillance Requirements are to be performed during shutdown, with the unit in mode 3 or higher. The footnote allows the particular surveillance to be performed during preplanned Preventative Maintenance (PM) activities that would result in the diesel generator being inoperable. The surveillance can be performed at that time as long as it does not increase the time the diesel generator is inoperable for the PM activity that is being performed. The footnote is only applicable at that time. The provision of the footnote shall not be utilized for operational convenience.

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."

In SURVEILLANCE 4.8.2.1.2.e, after the battery is returned to service (reconnected to and supplying its normal DC distribution center) following a performance discharge test (PDT), no discharge testing shall be done within 10 days on the other three batteries. This is a conservative measure to ensure the tested battery is fully charged. This restriction is an interim measure until the concern regarding recovered battery capacity immediately following recharging is resolved.

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-3 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-3 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Testing of these circuit breakers consists of injecting a current in excess of the breaker's nominal setpoint and measuring the response time. The measured response time is compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. 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.

Fuse testing is in accordance with IEEE Standard 242-1975. This program will detect any significant degradation of the fuses or improperly sized fuses. Safety is further assured by the "fail safe" nature of fuses, that is, if the fuse fails, the circuit will deenergize.

BASES

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 accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the minimum boron concentration value specified in the Core Operating Limits Report or greater includes a conservative uncertainty allowance of 50 ppm boron.

The Reactor Makeup Water Supply to the Chemical and Volume Control (NV) System is normally isolated during refueling to prevent diluting the Reactor Coolant System boron concentration. Isolation is normally accomplished by closing valve NV-250. However, isolation may be accomplished by closing valves NV-131, NV-140, NV-176, NV-468, NV-808, and either NV-132 or NV-1026, when it is necessary to makeup water to the Refueling Water Storage Tank during refueling operations.

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. In MODE 6 the Wide Range Neutron Flux Detectors (ENC) can be used as Source Range Neutron Flux Monitors. All of the LCO, ACTION, and SURVEILLANCE REQUIREMENTS of 3/4.9.2 must be met for the two Source Range Neutron Flux Monitors that are in use at any time.

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 accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY of the Reactor Building Containment Purge Exhaust System HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the Reactor Building Containment Purge Exhaust System HEPA filters and charcoal adsorbers and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis. The methyl iodide penetration test criteria for the carbon samples have been made more restrictive than required for the assumed iodine removal in the accident analysis because the humidity to be seen by the charcoal adsorbers may be greater than 70% under normal operating conditions.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: (1) manipulator cranes will be used for movement of 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 POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident 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 RHR capability. With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

To prevent vortexing in the suction of the RHR pumps, the flow rate requirements for the RHR system were lowered from 3000 gpm to 1000 gpm. A specific surveillance has been added to ensure the flow remains high enough to ensure the reactor coolant system temperature remains less than or equal to 140 degrees-F. The problems associated with vortexing and mid-loop operations is provided in Generic Letter 88-17, Loss of Decay Heat.

3/4.9.9 and 3/4.9.10 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 FUEL HANDLING VENTILATION EXHAUST SYSTEM

The limitations on the Fuel Handling Ventilation Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing. The methyl iodide penetration test criteria for the carbon samples have been made more restrictive than required for the assumed iodine removal in the accident analysis because the humidity to be seen by the charcoal adsorbers may be greater than 70% under normal operating conditions.

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE

The requirements for spent fuel pool boron concentration specified in Specification 3.9.12 ensure that a minimum boron concentration is maintained in the pool. The requirements for spent fuel assembly storage specified in Specification 3.9.13 ensure that the pool remains subcritical. The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence the design of the spent fuel storage racks is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the spent fuel pool fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 4) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 1 or Region 2. This could increase the reactivity of the spent fuel pool. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water.

Tables 3.9-1 through 3.9-5 allow for specific criticality analyses for fuel which does not meet the requirements for storage defined in these tables. These analyses would require using NRC approved methodology to ensure that $k_{eff} \leq 0.95$ with a 95 percent probability at a 95 percent confidence level as described in Section 9.1 of the FSAR. This option is intended to be used for fuel not included in previous criticality analyses. Fuel storage is still limited to the configurations defined in TS 3.9-13. The use of specific analyses for qualification of previously unanalyzed fuel includes, but is not

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE (Continued)

limited to, fuel assembly designs not previously analyzed which may be as a result of new fuel designs or fuel shipments from another facility. Another more likely, and expected use of this specific analysis provision would be to analyze movement and storage of individual fuel pins as a result of reconstitution activities.

In verifying the design criteria of $k_{eff} \leq 0.95$, the criticality analysis assumed the most conservative conditions, i.e. fuel of the maximum permissible reactivity for a given configuration. Since the data presented in Specification 3.9.13.a and 3.9.13.b represents the maximum reactivity requirements for acceptable storage, substitutions of less reactive components would also meet the $k_{eff} \leq 0.95$ criteria. Hence, any non-fuel component may be placed in a designated empty cell location. Likewise, an empty cell, or a non-fuel component may be substituted for any designated fuel assembly location. These, or other substitutions which will decrease the reactivity of a particular storage cell will only decrease the overall reactivity of the spent fuel storage pool.

If both restricted and unrestricted storage is used in Region 1, an additional criteria has been imposed to ensure that the boundary row between these two configurations would not locally increase the reactivity above the required limit. Likewise if checkerboard storage is used in Region 2, an additional restriction has been imposed on the boundaries of the checkerboard storage region to ensure that the reactivity would not increase above the required limit. No other restrictions on region interfaces are necessary.

For storage in Region 2 requiring loading pattern restrictions, (per Specifications 3.9.13.b.2 or 3.9.13.b.3) fuel may be stored in either the "cell" or "non-cell" locations. "Cell" locations are the areas inside the fabricated storage cells and "non-cell" locations are the storage locations created by arranging the fabricated storage cells in a checkerboard configuration. Hence the "non-cell" locations are the areas defined by the outside walls of the 4 adjacent "cell" locations.

The action statement applicable to fuel storage in the spent fuel pool requires that action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. Prior to the resumption of fuel movement, the requirements of the LCOs must be met. This requires restoring the soluble boron concentration and the correct fuel storage configuration to within the corresponding limits. This does not preclude movement of a fuel assembly to a safe position.

The surveillance requirements ensure that the requirements of the two LCOs are satisfied, namely boron concentration and fuel placement. The boron concentration in the spent fuel pool is verified to be greater than or equal to the minimum limit. The fuel assemblies are verified to meet the subcriticality requirement by meeting either the initial enrichment and burnup

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE (Continued)

requirements of Table 3.9-1 through 3.9-5, or by using NRC approved metnodology to ensure that $k_{eff} \leq 0.95$. By meeting either of these requirements, the analyzed accidents are fully addressed.

The fuel storage requirements and restrictions discussed here and applied in section 3.9.13 are based on a maximum allowable fuel enrichment of 4.75 weight% U-235. The enrichments listed in Tables 3.9-1 through 3.9-5 are nominal enrichments and include uncertainties to account for the tolerance on the as built enrichment. Hence the as built enrichments may exceed the enrichments listed in the tables by up to 0.05 weight% U-235. Qualifying burnups for enrichments not listed in the tables may be linearly interpolated between the enrichments provided. This is because the reactivity of an assembly varies linearly for small ranges of enrichment.

REFERENCES

1. "Regulatory Guide 1.13: Spent Fuel Storage Facility Design Basis", U.S. Nuclear Regulatory Commission, Office of Standards Development, Revision 1, December 1976.

2. "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations", American Nuclear Society, ANSI N210-1976/ANS-57.2, April 1976.

3. FSAR, Section 9.1.

4. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

BASES

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 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1. LIQUID EFFLUENTS

3/4.11.1.1 NOT USED

3/4.11.1.2 NOT USED

3/4.11.1.3 NOT USED

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.1 GASEOUS EFFLUENTS

3/4.11.2.1 NOT USED

3/4.11.2.2 NOT USED

3/4.11.2.3 NOT USED

3/4.11.2.4 NOT USED

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a Waste Gas System leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the WASTE GAS HOLDUP SYSTEM to assure that a release would be substantially below the dose guideline values of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

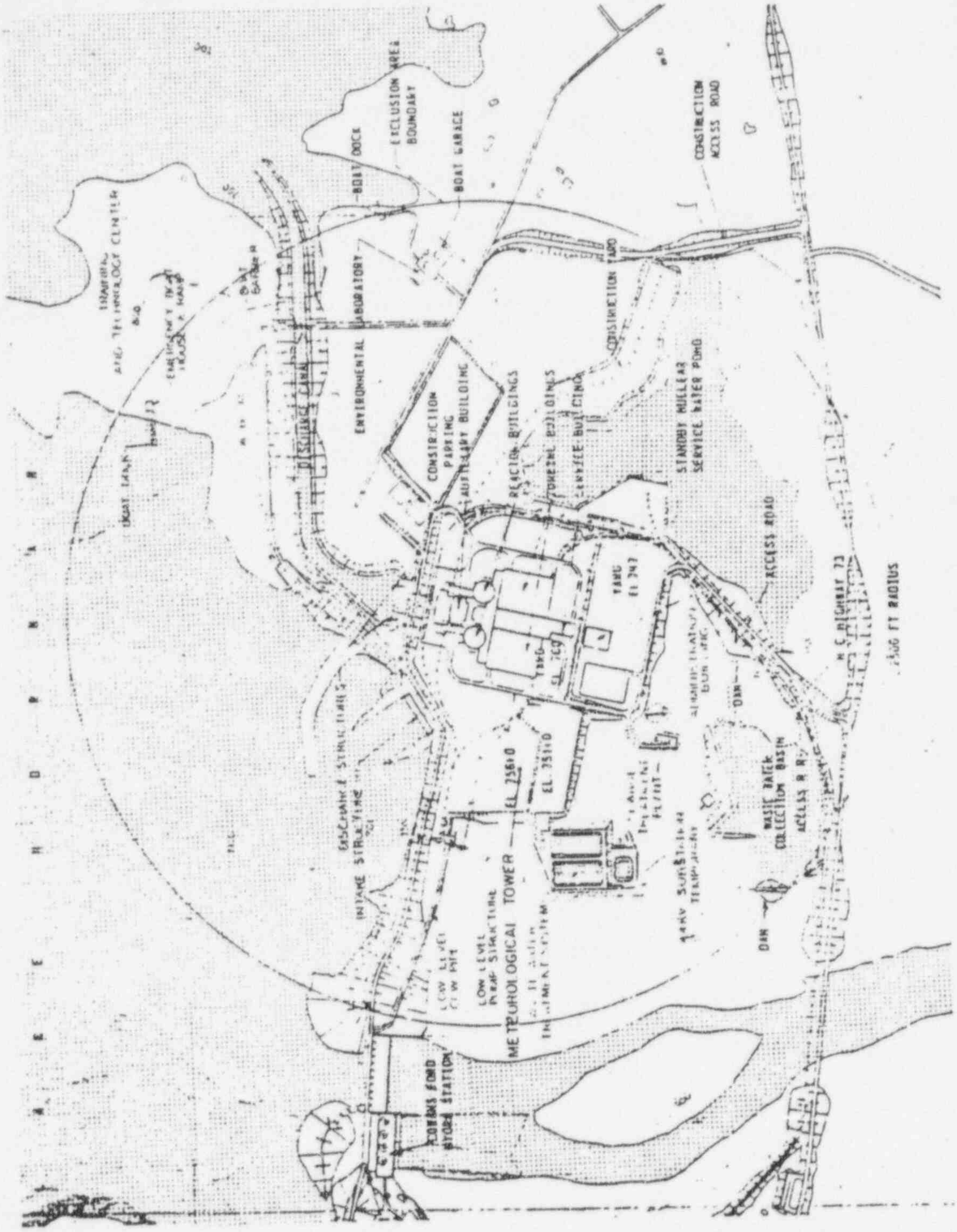
5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4. The definition of UNRESTRICTED AREA used in implementing the Radiological Effluent Technical Specifications has been expanded over that in 10 CFR Part 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR Part 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR Part 50.36a.



EXCLUSION AREA
McGUIRE NUCLEAR STATION

FIGURE 5.1-1

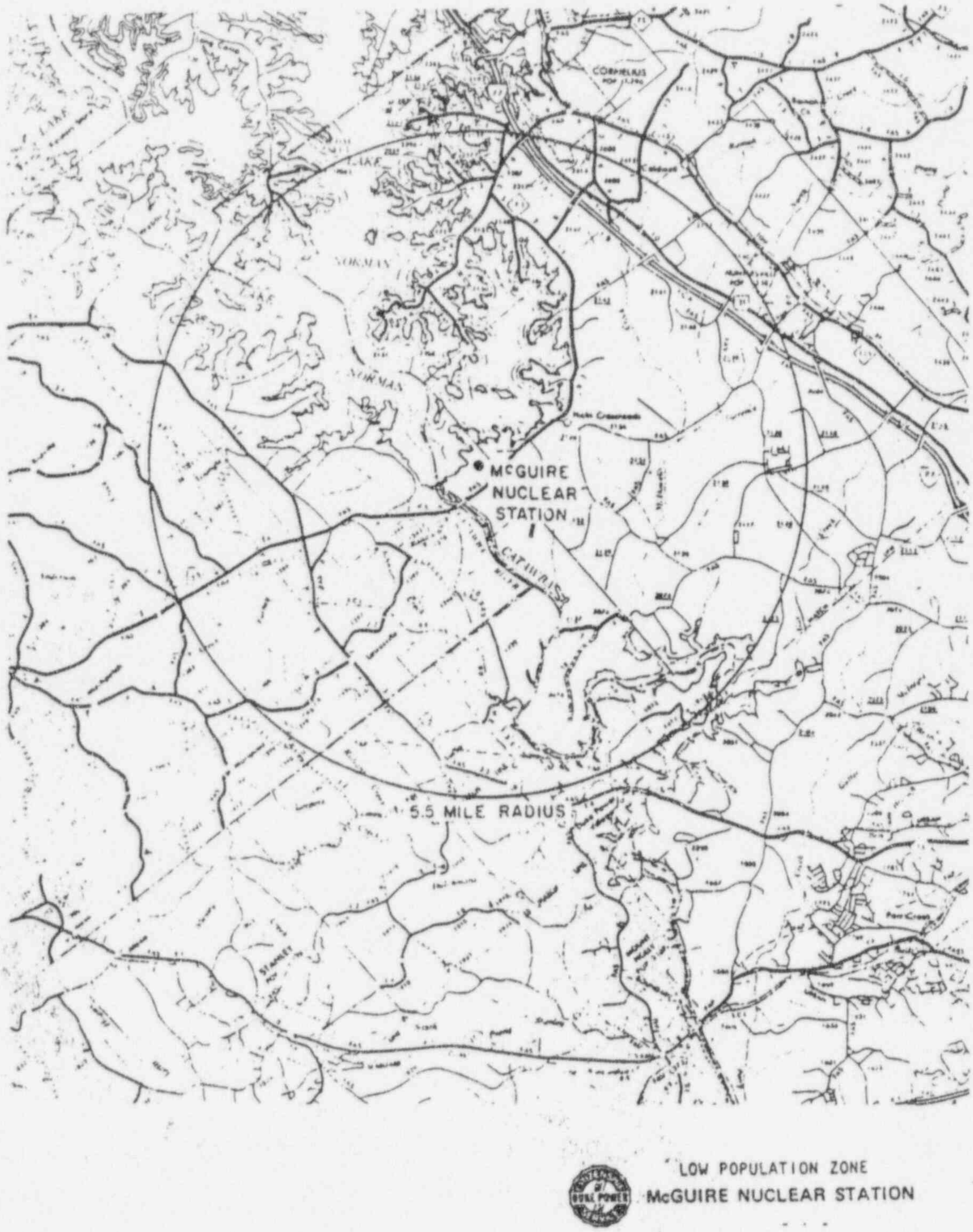
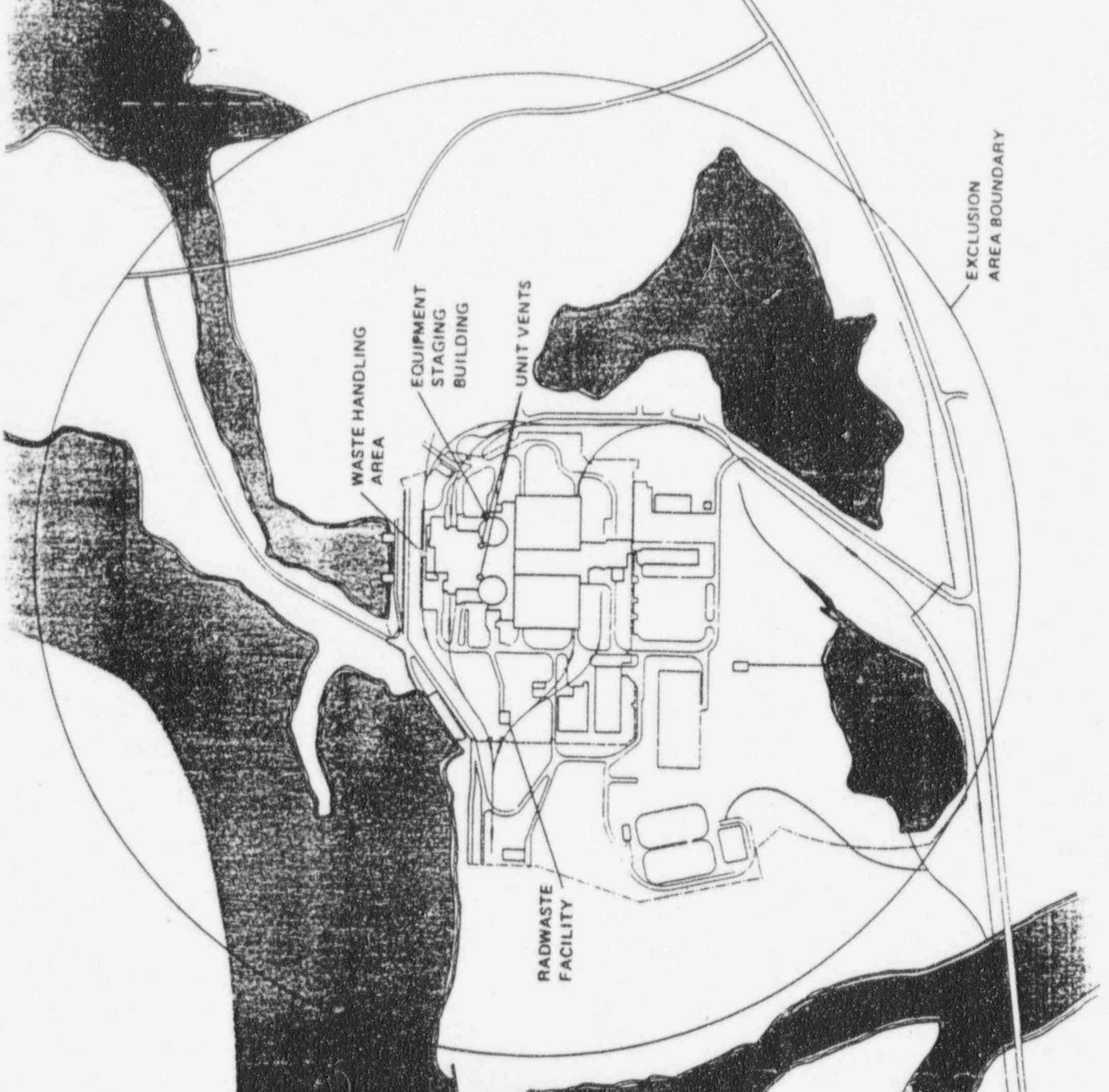
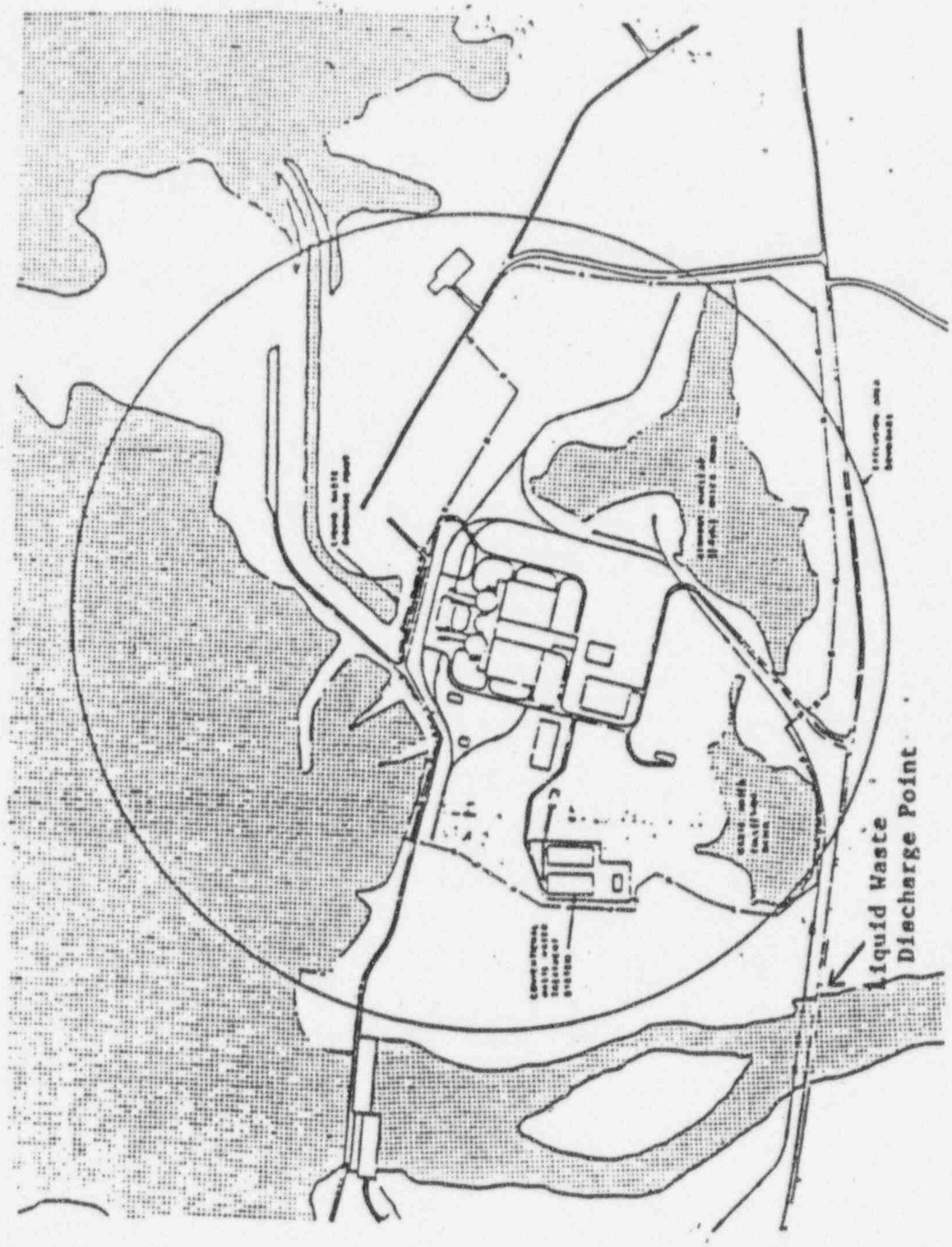

CORNELIUS
NORMAN
NORMAN
McGUIRE
NUCLEAR
STATION
CATAWBA
5.5 MILE RADIUS
LOW POPULATION ZONE
McGUIRE NUCLEAR STATION

FIGURE 5.1-2



SITE BOUNDARY FOR
GASEOUS EFFLUENTS
McGUIRE NUCLEAR STATION

FIGURE 5.1-3



SITE BOUNDARY FOR
LIQUID EFFLUENTS
McGUIRE NUCLEAR STATION

FIGURE 5.1-4

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel surrounded by a concrete containment.

5.2.1.1 CONTAINMENT VESSEL

a. Nominal inside diameter = 115 feet.

b. Nominal inside height = 169 feet.

c. Net free volume = 1.2×10^6 cubic feet.

d. Nominal thickness of vessel walls = 0.75 inch.

e. Nominal thickness of vessel dome = 0.6875 inch.

f. Nominal thickness of vessel bottom = 0.25 inch.

5.2.1.2 REACTOR BUILDING

a. Nominal annular space = 5 feet.

b. Annulus nominal volume = 427,000 cubic feet.

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 177 feet.

d. Nominal inside diameter = 125 feet.

e. Cylinder wall minimum thickness = 3 feet.

f. Dome minimum thickness = 2.25 feet.

g. Dome inside radius = 87 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 15.0 psig and a temperature of 250°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses

FUEL ASSEMBLIES Continued)

to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-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 for Unit 1 control rods shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,040 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) A nominal 10.4" center to center distance between fuel assemblies placed in Region 1; and

3) A nominal 9.125" center to center distance between fuel assemblies placed in Region 2.

b. The new fuel storage racks are designed and shall be maintained with:

1) $k_{eff} \leq 0.95$ if fully flooded with unborated water as described in Section 9.1 of the FSAR; and

2) $k_{eff} \leq 0.98$ if moderated by aqueous foam as described in Section 9.1 of the FSAR; and

3) A nominal 21" center to center distance between fuel assemblies placed in the storage racks.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745 ft. 7 in.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1463 fuel assemblies (286 spaces in Region 1 and 1177 spaces in Region 2).

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/hr and 200 cooldown cycles at ≤ 100°F/hr (pressurizer cooldown at ≤ 200°F/hr for 200°F ≤ T pressurizer ≤ 650°F).	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 551°F. Cooldown cycle - T_{avg} from ≥ 551°F to ≤ 200°F.
	80 loss of load cycles.	Without immediate Turbine or Reactor trip.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical power source supplying the Onsite Class 1E Distribution 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.
	200 large step decreases in load.	100% to 0% of RATED THERMAL POWER with steam dump.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	1 main reactor coolant pipe break.	Break in a reactor coolant pipe > 6 inches equivalent diameter.
	Operating Basis Earthquakes.	400 cycles - 20 earthquakes of 20 cycles each.
	50 leak tests.	Pressurized to 2500 psig.
	5 hydrostatic pressure tests.	Pressurized to 3107 psig.
Reactor Vessel	Operating Basis Earthquakes.	50 cycles.
Secondary Coolant System	1 steam line break.	Break in a steam line ≥ 6.0 inches equivalent diameter.
	5 hydrostatic pressure tests.	Pressurized to 1481 psig.

SECTION 6.0
ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice-President McGuire Nuclear Site shall be reissued to all McGuire Nuclear Site personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationship, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.

b. The Station Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President of McGuire Nuclear Site shall have responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The Senior Vice President Nuclear Generation Department will be the Senior Nuclear Executive and have corporate responsibility for overall nuclear safety.

e. The individuals who train the operating staff and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

6.2.2 The unit organization shall be as shown in the FSAR, Chapter 13, and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator for each unit shall be in the control room when fuel is in either reactor. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. A Radiation Protection Technician shall be on site when fuel is in either reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

e. Administrative procedures shall be developed and implemented to limit the working hours of station staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, radiation protection technicians, non-licensed operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 12 hour day with alternating 48 hour and 36 hour work week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 28 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time;

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time; and

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Station Manager or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Station Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3 or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
SS	1	1	1
SRO	1	none[b]	1
RO	3[a]	2[a]	3[a]
AO	3[a]	3[a]	3[a]
SM	1	none	1

S[illegible] - [illegible] Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary operator
SM - Shift Manager

[a] At least one of the required individuals must be assigned to the designated position for each unit.
[b] At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

TABLE 6.2-1 (Continued)

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Manager*) 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 or Operator license shall be designated to assume the control room command function.

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*On occasions when there is a need for both the Shift Supervisor and the SRO to be absent from the control room, the Shift Manager shall be allowed to assume the control room command function and serve as the SRO in the control room provided that: (1) the Shift Supervisor is available to return to the control room within 10 minutes, (2) the assumption of SRO duties by the Shift Manager be limited to periods not in excess of 15 minutes duration and a total time not to exceed 1 hour during any 8-hour shift, and (3) the Shift Manager has an SRO license on the unit.

6.2.3 McGUIRE SAFETY REVIEW GROUP

FUNCTION

6.2.3.1 The McGuire Safety Review Group (SRG) shall function to provide the review of plant design and operating experience for potential opportunities to improve plant safety; evaluation of plant operations and maintenance activities; and, to advise management on the overall quality and safety of plant operations. The SRG shall make recommendations for revised procedures, equipment modifications, or other means of improving plant safety to appropriate station/corporate management.

COMPOSITION

6.2.3.2 The SRG shall be composed of at least five individuals and at least three of these shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his/her field, at least 1 year of which experience shall be in the nuclear field.

The remaining individuals in the SRG shall have either (1) at least 5 years of nuclear experience and hold or have held a Senior Reactor Operator license; or (2) at least 8 years of professional level experience in his/her field, at least 5 years of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The SRG shall be responsible for:

a. Review of selected plant operating characteristics and other appropriate sources of plant design and operating experience information for awareness and incorporation into the performance of other duties.

b. Review of the effectiveness of corrective actions taken as a result of the evaluation of selected plant operating characteristics and other appropriate sources of plant design and operating experience information.

c. Review of selected programs, procedures, and plant activities, including maintenance, modification, operational problems, and operational analysis.

d. Surveillance of selected plant operations and maintenance activities to provide independent verification* that they are performed correctly and that human errors are reduced to as low as practicable.

e. Investigation of selected unusual events and other occurrences as assigned by Station Management or the Manager of Safety Assurance.

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*Not responsible for sign-off function.

6.2.3 McGUIRE SAFETY REVIEW GROUP (Continued)

AUTHORITY

6.2.3.4 The SRG shall report to and advise the Manager of Safety Assurance, on those areas of responsibility specified in Section 6.2.3.

RECORDS

6.2.3.5 Records of activities performed by the SRG shall be prepared and maintained for the life of the station. Summary reports of activities performed by the SRG shall be forwarded each calendar month to the Manager of Safety Assurance.

6.2.4 SHIFT MANAGER

6.2.4.1 The Shift Manager, whose functions include those of a Shift Technical Advisor, shall serve in an advisory capacity to the Shift Supervisor.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the SRG.

6.5 REVIEW AND AUDIT

6.5.1 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

Programs shall be established for the preparation, review, approval, and retention of documents required by the activities described in Specifications 6.5.1.1 through 6.5.1.11. Approvals shall be by the head of the appropriate site organization, the head of the appropriate station organization, the head of the appropriate site engineering organization, the head of the environmental organization, or an alternate as specified in other applicable regulatory documents or administrative controls.

6.5.1.1 Each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, shall be prepared by a knowledgeable individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/organization other than the individual/organization which prepared the procedure, or changes thereto. Procedures, or changes thereto, shall be approved in accordance with Specifications 6.8.2 and 6.8.3.

6.5.1.2 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a knowledgeable individual/organization. Each such modification shall be reviewed by an individual/organization other than the individual/organization which designed the modification.

6.5.1.3 Individuals responsible for reviews performed in accordance with Specifications 6.5.1.1 and 6.5.1.2 shall be members of the supervisory staff assigned to the site, previously designated by the Site Vice President to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated site review personnel.

6.5.1.4 Proposed changes to the Appendix A Technical Specifications shall be prepared by a knowledgeable individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/organization other than the individual/organization which prepared the proposed change.

6.5.1.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be prepared and approved in a manner identical to that of Specification 6.5.1.1. These proposed tests and experiments shall be reviewed by a knowledgeable individual/organization other than the individual/organization which prepared the proposed tests and experiments.

ACTIVITIES (Continued)

6.5.1.6 ALL REPORTABLE EVENTS and all violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be reviewed by a knowledgeable individual/organization other than the individual/organization which prepared the report.

6.5.1.7 Special reviews and investigations, and the preparation of reports thereon, shall be performed by a knowledgeable individual/organization.

5.5.1.8 A knowledgeable individual/organization shall review every unplanned onsite release of radioactive material to the environs and prepare reports covering evaluation, recommendations, and disposition of the corrective ACTION to prevent recurrence.

6.5.1.9 A knowledgeable individual/organization shall review changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and Radwaste Treatment Systems.

6.5.1.10 A knowledgeable individual/organization shall review the Fire Protection Program and implementing procedures.

6.5.1.11 Reports documenting each of the activities performed under Specifications 6.5.1.1 through 6.5.1.10 shall be maintained.

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB)

FUNCTION

6.5.2.1 The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

FUNCTION (Continued)

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Administrative control and quality assurance practices.

ORGANIZATION

6.5.2.2 The Director, members and alternate members of the NSRB shall be appointed in writing by the Senior Vice President, Nuclear Generation and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of 5 years technical experience, of which a minimum of 3 years shall be in one or more areas given in Specification 6.5.2.1. In special cases, candidates for appointment without an academic degree in engineering or physical science may be qualified with a minimum of ten years experience in one of the areas in Specification 6.5.2.1. No more than two alternates shall participate as voting members in NSRB activities at any one time.

6.5.2.3 The NSRB shall be composed of at least five members, including the Director. Members of the NSRB may be from the Nuclear Generation Department, from other departments within the Company, or from external to the Company. A maximum of one member of the NSRB may be from the McGuire Nuclear Site staff.

6.5.2.4 Consultants shall be utilized as determined by the NSRB Director to provide expert advice to the NSRB.

6.5.2.5 Staff assistance may be provided to the NSRB in order to promote the proper, timely, and expeditious performance of its functions.

6.5.2.6 The NSRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least twice per year thereafter.

6.5.2.7 The quorum of the NSRB necessary for the performance of the NSRB review and audit functions of these Technical Specifications shall consist of the Director, or designated alternate, and at least four other NSRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of McGuire Nuclear Station.

REVIEW

6.5.2.8 The NSRB shall review:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

d. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

e. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

f. All REPORTABLE EVENTS;

g. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems or components that could affect nuclear safety;

h. Quality Assurance Program audits relating to station operations and actions taken in response to these audits; and

i. Reports of activities performed under the provisions of Specifications 6.5.1.1 through 6.5.1.10.

AUDITS

6.5.2.9 Audits of site activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions;

b. The performance, training, and qualifications of the entire station staff;

AUDITS (Continued)

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50;

e. The Emergency Plan and implementing procedures;

f. The Security Plan and implementing procedures;

g. The Facility Fire Protection programmatic controls including the implementing procedures;

h. The fire protection equipment and program implementation utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

i. The Radiological Environmental Monitoring Program and the results thereof;

j. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures;

k. The PROCESS CONTROL PROGRAM and implementing procedures for SOLIDIFICATION of radioactive wastes;

l. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring and;

m. Any other area of site operation considered appropriate by the NSRB or the Senior Vice President, Nuclear Generation.

AUTHORITY

6.5.2.10 The NSRB shall report to and advise the Senior Vice President, Nuclear Generation on those areas of responsibility specified in Specifications 6.5.2.8 and 6.5.2.9.

RECORDS

6.5.2.11 Records of NSRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRB meeting shall be prepared, approved, and forwarded to the Senior Vice President, Nuclear Generation and to the Site Vice President, within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.8 above, shall be prepared, approved and forwarded to the Senior Vice President, Nuclear Generation, and to the Site Vice President, within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.9 above, shall be forwarded to the Senior Vice President, Nuclear Generation and to the Site Vice President, and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the Station Manager; or for the Station Manager by: (1) the Operations Superintendent, (2) the Maintenance Superintendents, or (3) the Work Control Superintendent, as previously designated by the Station Manager, and the results of the review shall be submitted to the NSRB and the Site Vice President.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice President, and the NSRB shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Operations Superintendent and the Station Manager. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence;

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Site Vice President, within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The applicable procedures required to implement the requirements of NUREG-0737;

c. Deleted

d. Deleted

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation; and

g. Quality Assurance Program for effluent and environmental monitoring.

h. Technical Review and Control Program implementation.

i. Fire Protection Program implementation.

j. Plant Operations Review Committee implementation.

k. Commitments contained in FSAR Chapter 16.0

6.8.2 Each procedure of Specification 6.8.1 above, and changes thereto, shall be reviewed and approved by an appropriate division manager, superintendent/manager, or one of their designated direct reports prior to implementation and shall be reviewed periodically as set forth in administrative procedures. For procedures which implement offsite environmental, technical, and laboratory activities, the above review and approval may be performed by the General Manager, Environmental Services or designee.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above 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 approved by an appropriate division manager, superintendent/manager, or one of their designated direct reports within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Reactor 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 RHR, Boron Recycle, Refueling Water, Liquid Waste, Waste Gas, Safety Injection, Chemical and Volume Control, Containment Spray, and Nuclear Sampling. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) 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,

PROCEDURES AND PROGRAMS (Continued)

4) Procedures for the recording and management of data,

5) Procedures defining corrective actions for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1) Training of personnel, and

2) Procedures for monitoring.

e. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines, and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

f. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in FSAR Chapter 16, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and set-point determination in accordance with the methodology in the Offsite Dose Calculation Manual (ODCM),

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 times 10 CFR Part 20.1001-20.2401, Appendix B, Table 2, Column 2,

PROCEDURES AND PROGRAMS (Continued)

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following:

   a. For noble gases: Less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

   b. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrems/yr to any organ.

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

PROCEDURES AND PROGRAMS (Continued)

g. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in FSAR Chapter 16, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant STARTUP and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the License involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

STARTUP REPORT (Continued)

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in License conditions based on other commitments shall be included in this report.

6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the STARTUP test program, or (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 [1/]

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Annual Reports shall include the activities of the unit as described below:

a. Personnel Exposures

Reports required on an annual basis shall include 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, 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.

[1/] 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[1/]

b. Primary Coolant Specific Activity

Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be include: 1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; 2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded 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 system flow history starting 48 hours prior to the first sample in which the limit was exceeded; 4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and 5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 Routine 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 report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

1/ 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.

* A single submittal may be made for a multiple unit station.

** 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.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference limits, target band*, and APL^{ND}* for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, F^{RTP}_{Q}, K(Z), W(Z)**, APL^{ND}** and $W(Z)_{BL}$** for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F^{L}_{\Delta H}$***, or $F^{RTP}_{\Delta H}$****, and Power Factor Multiplier, $MF_{\Delta H}$****, limits for Specification 3/4.2.3.
7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 2.2.1.
8. Boric Acid Storage System and Refueling Water Storage Tank volume and boron concentration limits for Specifications 3/4.1.2.5 and 3/4.1.2.6.
9. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3/4.5.1 and 3/4.5.5.
10. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.
11. Spent fuel pool boron concentration limits for Specification 3/4.9.12.

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* Reference 5 is not applicable to target band and APL^{ND}.

** References 4 and 5 are not applicable to W(Z), APL^{ND}, and $W(Z)_{BL}$.

*** Reference 1 is not applicable to $F^{L}_{\Delta H}$.

**** Reference 5 is not applicable to $F^{RTP}_{\Delta H}$ and $MF_{\Delta H}$.

CORE OPERATING LIMITS REPORT

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987 (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," SER dated January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

5. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990 (DPC Proprietary).

   (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NE-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3/4.9.12 - Spent Fuel Pool Boron Concentration.)

CORE OPERATING LIMITS REPORT

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, "Thermal-Hydraulic Transient Analysis Methodology," August 1994.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November, 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H(X,Y)$.)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation setpoints).

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

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.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

RECORD RETENTION (Continued)

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 reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

j. Records of meetings of the NSRB and reports required by Specification 6.5.1.11;

k. Records of the service lives of all snubbers including the date at which the service life commences and associated installation and maintenance records;

l. Records of secondary water sampling and water quality; and

m. 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.

n. Records of reviews performed for changes made to the ODCM and the PCP.

6.10.3 Records of quality assurance activities required by the QA Manual shall be retained for a period of time required by ANSI N45.2.9-1974.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mrem/hr at 45 CM (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Radiation Protection Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mrem/hr provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

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 level in the area has 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 Radiation Protection Manager 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 mrem/hr 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 area 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 areas accessible to personnel with radiation levels greater than 1000 mrem/hr* that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, that area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2n. This documentation shall contain:

*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (PCP) (Continued)

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective upon review and acceptance by the Station Manager and a qualified individual/organization.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2n. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective upon review and acceptance by the station manager and a qualified individual/organization.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee initiated major changes to the Radioactive Waste Systems (liquid, gaseous and solid)

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Station Manager. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the Station Manager or the Chemistry Manager.

b. Shall become effective upon review and acceptance by a qualified individual/organization.

*Licensees may choose to submit the information called for in this specification as part of the annual FSAR update.