

January 3, 1995

Mr. John F. Opeka
Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, CT 06141-0270

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M89653)

Dear Mr. Opeka:

The Commission has issued the enclosed Amendment No. 100 to Facility Operating License No. NPF-49 for the Millstone Nuclear Power Station, Unit No. 3, in response to your application dated June 2, 1994, as supplemented August 25, 1994.

The amendment revises the Technical Specifications (TS) to remove expired one-time extensions of surveillance, removes an obsolete definition of charging pump operability, and incorporates 11 line item improvements in accordance with the guidance provided in Generic Letter 93-05. Several editorial changes have been made to renumber TS pages and delete the blank pages from the TS.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Vernon L. Rooney, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 100 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

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* No legal objection, w/comments on pg. 103 of SER.

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DOCUMENT NAME: G:\ROONEY\M89653.AMD

OFFICE	LA:PDI-4	PM:PDI-4	PM:PDI-4	D:PDI-4	OGC <i>Raman*</i>
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DATE	12/12/94	12/5/94	12/6/94	12/17/94	12/23/94

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* No legal objection, w/ comments on pg 103 of SER.



UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

January 3, 1995

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Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
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Sincerely,

A handwritten signature in black ink, appearing to read "V. Rooney", written over a horizontal line.

Vernon L. Rooney, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 100 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

Mr. John F. Opeka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit 3

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

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 100
License No. NPF-49

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee), dated June 2, 1994, as supplemented August 25, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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P PDR

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 100, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Phillip F. McKee, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: January 3, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 100

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove

ii
ix
x
xv
1-7
3/4 2-20
3/4 2-23
3/4 3-1
3/4 3-10
3/4 3-11
3/4 3-12
3/4 3-14
3/4 3-16
3/4 3-36
3/4 3-37
3/4 3-38
3/4 3-39
3/4 3-40
3/4 3-41
3/4 3-45
3/4 3-49
3/4 3-53
3/4 3-58
3/4 3-62
3/4 3-63
3/4 3-68
3/4 4-11
3/4 4-16
3/4 4-21
3/4 4-23
3/4 4-39
3/4 4-43
3/4 5-1
3/4 5-2
3/4 5-4
3/4 6-3
3/4 6-12
3/4 6-13
3/4 6-16 thru 3/4 6-42

Insert

ii
ix
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1-7
3/4 2-20
3/4 2-23
3/4 3-1
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3/4 3-11
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3/4 3-39
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3/4 3-45
3/4 3-49
3/4 3-53
3/4 3-58
3/4 3-62
3/4 3-63
3/4 3-68
3/4 4-11
3/4 4-16
3/4 4-21
3/4 4-23
3/4 4-39
3/4 4-43
3/4 5-1
3/4 5-2
3/4 5-4
3/4 6-3
3/4 6-12
3/4 6-13
3/4 6-16 thru 3/4 6-23

(continued)

Remove

3/4 7-4
3/4 7-5
3/4 7-23
3/4 7-26a thru 3/4 7-47
3/4 8-5
3/4 8-7
3/4 8-12
B 3/4 5-2
B 3/4 7-7
B 3/4 7-8

Insert

3/4 7-4
3/4 7-5
3/4 7-23
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DEFINITIONS

VENTING

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

SPENT FUEL POOL STORAGE PATTERNS:

1.40 Region I spent fuel racks contain a cell blocking device in every 4th location for criticality control. This 4th location will be referred to as the blocked location. A STORAGE PATTERN refers to the blocked location and all adjacent and diagonal Region I cell locations surrounding the blocked location. Boundary configuration between Region I and Region II must have cell blockers positioned in the outermost row of the Region I perimeter, as shown in Figure 3.9-2.

1.41 Region II contains no cell blockers.

CORE OPERATING LIMITS REPORT (COLR)

1.42 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 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

ALLOWED POWER LEVEL

1.43 APLND is the minimum allowable nuclear design power level for base load operation and is specified in the COLR.

1.44 APL^{BL} is the maximum allowable power level when transitioning into base load operation.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

- b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that $F_{\Delta H}^N$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.
- c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}^N$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:
 - 1. A nominal 50% of RATED THERMAL POWER,
 - 2. A nominal 75% of RATED THERMAL POWER, and
 - 3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

- 4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.
- 4.2.3.1.2 RCS total flow rate and $F_{\Delta H}^N$ shall be determined to be within the acceptable range:
 - a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and
 - b. At least once per 31 Effective Full Power Days.
- 4.2.3.1.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F_{\Delta H}^N$, obtained per Specification 4.2.3.1.2, is assumed to exist.
- 4.2.3.1.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

- b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that $F_{\Delta H}^N$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.
- c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}^N$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:
 1. A nominal 32% of RATED THERMAL POWER, and
 2. A nominal 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

- 4.2.3.2.1 The provisions of Specification 4.0.4 are not applicable.
- 4.2.3.2.2 RCS total flow rate and $F_{\Delta H}^N$ shall be determined to be within the acceptable range at least once per 31 Effective Full Power Days.
- 4.2.3.2.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F_{\Delta H}^N$, obtained per Specification 4.2.3.2.2, is assumed to exist.
- 4.2.3.2.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.
- 4.2.3.2.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.
- 4.2.3.2.6 If the feedwater venturis are not inspected at least once per 18 months, an additional 0.1% will be added to the total RCS flow measurement uncertainty.

3/4.3 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.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
5. Intermediate Range	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1), Q(9)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	S	R	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(18)	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	Q(18)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	S	R	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
20. Three Loop Operation Bypass Circuitry	N.A.	N.A.	N.A.	R	N.A.	1, 2
21. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	M(15) R(16)	N.A.	1, 2, 3*, 4*, 5*
22. Shutdown Margin Monitor	N.A.	N.A.	Q(19)	N.A.	N.A.	3, 4, 5

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

- (10) Setpoint verification is not applicable.
- (11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.
- (12) (not used)
- (13) Reactor Coolant Pump Shaft Speed Sensor may be excluded from CHANNEL CALIBRATION.
- (14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).
- (15) Local manual shunt trip prior to placing breaker in service.
- (16) Automatic undervoltage trip.
- (17) (not used).
- (18) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 should be reviewed for applicability.
- (19) Quarterly surveillance shall include verification that the Shutdown Margin Monitor is set per the CORE OPERATING LIMITS REPORT (COLR).

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

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

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME* of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel (to include input relays to both trains) 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.

*The provisions of Specification 4.0.4 are not applicable for response time testing of steam line isolation for entry into MODE 4 and MODE 3 and turbine driven auxiliary feedwater pump for entry into MODE 3.

TABLE 4.3-2

**ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS**

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-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-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
4. Steam Line Isolation								
a. Manual Initiation								
1) Individual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) System	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
4. Steam Line Isolation (Continued)								
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-2	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High	S	R	Q	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Safety Injection Actuation Logic	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
d. T _{ave} Low Coincident with Reactor Trip (P-4)	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power	See Item 8. below for all Loss of Power Surveillance.							
f. Containment Depressurization Actuation (CDA)	See Item 2. above for all CDA Surveillance Requirements.							
7. Control Building Isolation								
a. Manual Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
d. Containment Pressure--High-1	S	R	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

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
7. Control Building Isolation (Continued)								
e. Control Building Inlet Ventilation Radiation	S	R	Q	N.A.	N.A.	N.A.	N.A.	All
8. Loss of Power								
a. 4 kV Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	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
10. Emergency Generator Load Sequencer	N.A.	N.A.	N.A.	N.A.	Q(1, 2)	N.A.	N.A.	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

TABLE NOTATION

- (1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
- (2) This surveillance may be performed continuously by the emergency generator load sequencer auto test system as long as the EGLS auto test system is demonstrated operable by the performance of an ACTUATION LOGIC TEST at least once per 92 days.
- (3) On a monthly basis, a loss of voltage condition will be initiated at each undervoltage monitoring relay to verify individual relay operation. Setpoint verification and actuation of the associated logic and alarm relays will be performed as part of the channel calibration required once per 18 months.

TABLE 4.3-3

**RADIATION MONITORING INSTRUMENTATION FOR PLANT
OPERATIONS SURVEILLANCE REQUIREMENTS**

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Containment				
a. Containment Area Purge and Exhaust Isolation	S	R	Q	5, 6
b. RCS Leakage Detection				
1) Particulate Radio- activity	S	R	Q	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	Q	1, 2, 3, 4
2. Fuel Storage Pool Area Monitors				
a. Radiation Level	S	R	Q	*

TABLE NOTATIONS

* With fuel in the fuel storage pool area.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.
- b. With one or more Remote Shutdown Instrumentation transfer switches, power, or control circuits inoperable, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.
- c. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 4.3-6

**REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS**

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>
1. Reactor Trip Breaker Indication	M	N.A.
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Pressure	M	R
5. Steam Generator Water Level	M	R
6. Auxiliary Feedwater Flow Rate	M	R
7. Loop Hot Leg Temperature	M	R
8. Loop Cold Leg Temperature	M	R
9. Reactor Coolant System Pressure (Wide Range)	M	R
10. DWST Level	M	R
11. RWST Level	M	R
12. Containment Pressure	M	R
13. Emergency Bus Voltmeters	M	R
14. Source Range Count Rate	M*	R
15. Intermediate Range Amps	M	R
16. Boric Acid Tank Level	M	R

* When below P-6 (intermediate range neutron flux interlock setpoint).

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>
1. Containment Pressure		
a. Normal Range	M	R
b. Extended Range	M	R
2. Reactor Coolant Outlet Temperature - T _{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T _{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Demineralized Water Storage Tank Water Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. Containment Water Level (Wide Range)	M	R
14. Core Exit Thermocouples	M	R
15. DELETED		

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

** Electronic calibration from the ICC cabinets only.

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Amendment No. 75, 79, 100

INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

- a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.
- b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

- a. A CHANNEL CHECK at least once per 24 hours,
- b. An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and
- c. A CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% (1656 cubic feet), and at least two groups of pressurizer heaters supplied by emergency power each having a capacity of at least 175 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With only one group of pressurizer heaters supplied by emergency power OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

REACTOR COOLANT SYSTEM

STEAM GENERATORS

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. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;
- b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and
- c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:
 - 1) Primary-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or
 - 2) A seismic occurrence greater than the Operating Basis Earthquake, or
 - 3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or
 - 4) A main steam line or feedwater line break.

REACTOR COOLANT SYSTEM

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. Either the Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and
- b. The Containment Drain Sump Level or Pumped Capacity Monitoring System

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

ACTION:

- a. With both the Containment Atmosphere Gaseous and Particulate Radioactivity Monitors INOPERABLE, operation may continue for up to 30 days provided the Containment Drain Sump Level or Pumped Capacity Monitoring System is OPERABLE and gaseous grab samples of the containment atmosphere are obtained at least once per 12 hours and analyzed for gross noble gas activity within the subsequent 2 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 Drain Sump Level or Pumped Capacity Monitoring System INOPERABLE, operation may continue for up to 30 days provided either the Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System is OPERABLE; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

- 4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:
- a. Containment Atmosphere Gaseous and Particulate Radioactivity Monitoring Systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and
 - b. Containment Drain Sump Level and Pumped Capacity Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

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

- a. Monitoring the containment atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours;
- b. Monitoring the containment drain sump inventory and discharge at least once per 12 hours;
- c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2250 ± 20 psia at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;
- d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and
- e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

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

- a. At least once per 18 months,
- b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,
- c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,
- d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, and
- e. As outlined in the ASME Code, Section XI, paragraph IWV-3427(b).

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

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

- e. In the event the PORVs, the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.
- f. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

- a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;
- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and
- c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

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

- a. For RHR suction relief valve 3RHS*RV8708A, by verifying at least once per 12 hours that 3RHS*MV8701A and 3RHS*MV8701C are open;
- b. For RHR suction relief valve 3RHS*RV8708B, by verifying at least once per 12 hours that 3RHS*MV8702B and 3RHS*MV8702C are open; and
- c. Testing pursuant to Specification 4.0.5.

REACTOR COOLANT SYST

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of two parallel trains with two valves in series powered from emergency busses shall be OPERABLE and the vent closed* at each of the following locations:

- a. Reactor vessel head, and
- b. Pressurizer steam space.

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

ACTION:

- a. With one train of the reactor vessel head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable train is maintained closed with power removed from the valve actuators of all valves in the inoperable train; restore the inoperable train 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 trains of the reactor vessel head vent paths inoperable; maintain both trains closed with power removed from the valve actuators of all valves in the inoperable trains, and restore at least one of the trains to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With any valve(s) of the pressurizer steam space vent path inoperable in MODES 1, 2, or 3, follow the ACTION requirements of Specification 3.4.4.
- d. With any valve(s) of the pressurizer steam space vent path inoperable in MODE 4, follow the ACTION requirements of Specification 3.4.9.3.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each train of the reactor vessel head vent path isolation valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per COLD SHUTDOWN, if not performed within the previous 92 days, by operating the valve through one complete cycle of full travel from the control room.

*For an OPERABLE vent path using a power-operated relief valve (PORV) as the vent path, the PORV block valve is not required to be closed.

3/4.5 EMERGENCY COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

- 3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:
- a. The isolation valve open and power removed,
 - b. A contained borated water volume of between 6618 and 7030 gallons,
 - c. A boron concentration of between 2600 and 2900 ppm, and
 - d. A nitrogen cover-pressure of between 636 and 694 psia.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

- a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.
- b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

- 4.5.1 Each accumulator shall be demonstrated OPERABLE:
- a. At least once per 12 hours by:
 - 1) Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and
 - 2) Verifying that each accumulator isolation valve is open.
 - b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution. This surveillance is not required when the volume increase makeup source is the RWST.

*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

EMERGENCY CORE COOLING SYSTEMS

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:

<u>Valve Number</u>	<u>Valve Function</u>	<u>Valve Position</u>
3SIH*MV8806	RWST Supply to SI Pumps	OPEN
3SIH*MV8802A	SI Pump A to Hot Leg Injection	CLOSED
3SIH*MV8802B	SI Pump B to Hot Leg Injection	CLOSED
3SIH*MV8835	SI Cold Leg Master Isolation	OPEN
3SIH*MV8813	SI Pump Master Miniflow Isolation	OPEN
3SIL*MV8840	RHR to Hot Leg Injection	CLOSED
3SIL*MV8809A	RHR Pump A to Cold Leg Injection	OPEN
3SIL*MV8809B	RHR Pump B to Cold Leg Injection	OPEN

- b. At least once per 31 days by:

- 1) Verifying that the ECCS piping, except for the RSS pump, heat exchanger and associated piping, is full of water by venting the ECCS pump casings and accessible discharge piping high points, and
- 2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

- 1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and
- 2) At least once daily of the areas affected (during each day) within containment by containment entry and during the final 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 390 psia the interlocks prevent the valves from being opened.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c. The accuracy of each Type A test shall be verified by a supplemental test which:
 - 1) Confirms the accuracy of the test by verifying that the supplemental test results, L_o , minus the sum of the Type A and the superimposed leak, L_o , is equal to or less than $0.25 L_o$;
 - 2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and
 - 3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between $0.75 L_o$ and $1.25 L_o$.
- d. Type B and C tests shall be conducted with gas at P_o , 53.27 psia (38.57 psig), at intervals no greater than 24 months except for tests involving:
 - 1) Air locks
- e. The combined bypass leakage rate shall be determined to be less than or equal to $0.042 L_o$ 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_o , 53.27 psig (38.57 psig), during each Type A test;
- f. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;
- g. Purge supply and exhaust isolation valves shall be demonstrated OPERABLE by the requirements of Specifications 4.6.3.2.c and 4.9.9.
- h. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Quench Spray subsystems shall be OPERABLE.

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

ACTION:

With one Containment Quench Spray subsystem inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Quench Spray subsystem shall be demonstrated OPERABLE:

- a. At least once per 31 days:
 - 1) Verifying that each valve (manual, power operated, or automatic) in the flow path is not locked, sealed, or otherwise secured in position, is in its correct position; and
 - 2) Verifying the temperature of the borated water in the refueling water storage tank is between 40°F and 50°F.
- b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 114 psid when tested pursuant to Specification 4.0.5;
- c. At least once per 18 months during shutdown, by:
 - 1) Verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal, and
 - 2) Verifying that each spray pump starts automatically on a CDA test signal.
- d. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT SYSTEMS

RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent Recirculation Spray Systems shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each Recirculation Spray System shall be demonstrated OPERABLE:

- a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is not locked, sealed, or otherwise secured in position, is in its correct position;
- b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 130 psid when tested pursuant to Specification 4.0.5;
- c. At least once per 18 months by verifying that on a CDA test signal, each recirculation spray pump starts automatically after a 660 ± 20 second delay;
- d. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal; and
- e. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.
- b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.
- c. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

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 92 days, and at least once each refueling interval by performing a CHANNEL CALIBRATION using sample gas containing:

- a. One volume percent hydrogen, balance nitrogen, and
- b. Four volume percent hydrogen, balance nitrogen.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE at least once each refueling interval by:

- a. Verifying during a Hydrogen Recombiner System functional test that the minimum reaction chamber gas temperature increases to greater than or equal to 700°F within 90 minutes and is maintained for at least 2 hours and that the purge blower operates for 15 minutes.
- b. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,
- c. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.),
- d. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than 10,000 ohms, and
- e. Verifying during a recombiner system functional test using containment atmospheric air at an acceptable flow rate as determined in Section 4.6.4.2.f that the gas temperature increases to greater than or equal to 1100°F within 5 hours and is maintained for at least 4 hours.
- f. Verifying during a recombiner system functional test using containment atmospheric air that the blower would be capable of delivering at least 41.52 scfm at containment conditions of 12.47 psia and 130°F.

CONTAINMENT SYSTEMS

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the inside or outside isolation valves in the steam jet air ejector suction line not closed, restore the valve to the closed position within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Supplementary Leak Collection and Release Systems shall be OPERABLE with each system comprised of:

- a. one OPERABLE filter and fan, and
- b. one OPERABLE Auxiliary Building Filter System as defined in Specification 3.7.9.

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

ACTION:

With one Supplementary Leak Collection and Release System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Supplementary Leak Collection and Release System shall be demonstrated OPERABLE:

- a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying a system flow rate of 7600 cfm to 9800 cfm and that the system operates for at least 10 continuous hours with the heaters operating.
- b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:
 - 1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 7600 cfm to 9800 cfm;

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978,* for a methyl iodide penetration of less than 0.175%; and
 - 3) Verifying a system flow rate of 7600 cfm to 9800 cfm during system operation when tested in accordance with ANSI N510-1980.
- c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978,* for a methyl iodide penetration of less than 0.175%:
- d. At least once per 18 months by:
- 1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.25 inches Water Gauge while operating the system at a flow rate of 7600 cfm to 9800 cfm,
 - 2) Verifying that the system starts on a Safety Injection test signal,
 - 3) Verifying that each system produces a negative pressure of greater than or equal to 0.4 inch Water Gauge in the Auxiliary Building at 24'6" elevation within 120 seconds after a start signal, and
 - 4) Verifying that the heaters dissipate 50 ± 5 kW when tested in accordance with ANSI N510-1980.

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

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 7600 cfm to 9800 cfm; and
- f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 7600 cfm to 9800 cfm.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT BOUNDARY

LIMITING CONDITION FOR OPERATION

3.6.6.2 SECONDARY CONTAINMENT BOUNDARY shall be maintained.

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

ACTION:

Without SECONDARY CONTAINMENT BOUNDARY, restore SECONDARY CONTAINMENT BOUNDARY 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 REQUIREMENT

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

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT BOUNDARY STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the structural integrity of the SECONDARY CONTAINMENT BOUNDARY not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENT

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

PLANT SYSTEMS

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. Entry into an OPERATIONAL MODE pursuant to Specification 3.0.4 is not permitted with three auxiliary feedwater pumps inoperable.

SURVEILLANCE REQUIREMENTS

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

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. At least once per 92 days on a STAGGERED TEST BASIS 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; and
 - 2) Verifying that each auxiliary feedwater control and isolation valve in the flow path is in the fully open position when above 10% RATED THERMAL POWER.

- c. At least once per 18 months during shutdown by verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal. For the steam turbine-driven auxiliary feedwater pump, the provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying flow to each steam generator.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.10.f. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan for each type shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

- 1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.10f., an additional 5% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

**TABLE 4.7-2
SNUBBER VISUAL INSPECTION INTERVAL**

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend 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. These 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 sizes 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 included a fractional value of unacceptable snubbers as determined 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 no greater than 48 months.

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 interval 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.

TABLE 4.7-2
SNUBBER VISUAL INSPECTION INTERVAL

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

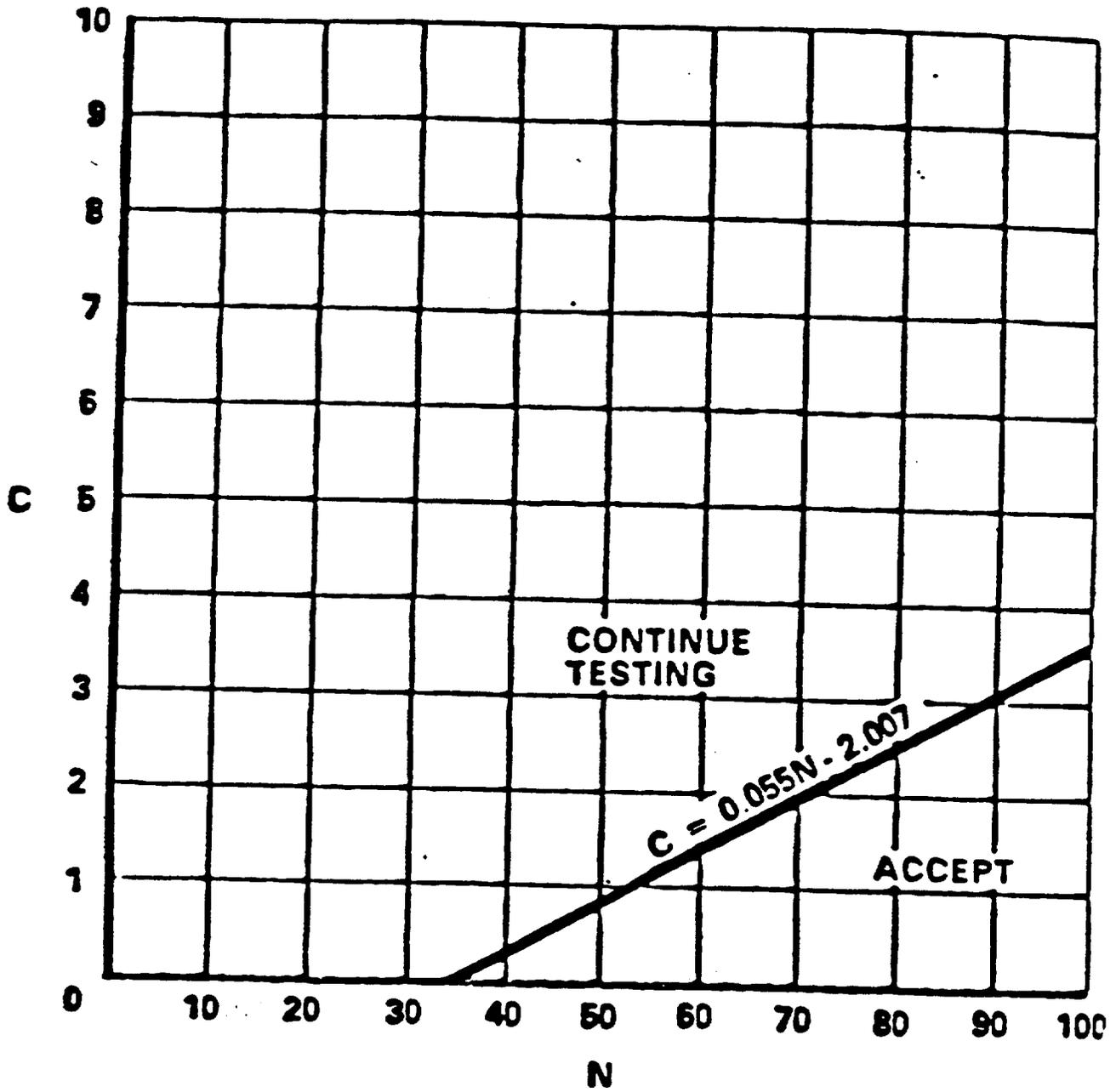


FIGURE 4.7-1
 SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

PLANT SYSTEMS

3/4.7.11 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.11 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

- a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:
 1. Decontaminate and repair the sealed source, or
 2. Dispose of the sealed source in accordance with Commission Regulations.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

- a. The licensee, or
- b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.11.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

- a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:
 - 1) With a half-life greater than 30 days (excluding Hydrogen 3), and
 - 2) In any form other than gas.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and
 - c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.
- 4.7.11.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

PLANT SYSTEMS

3/4.7.14 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

- a. By less than 20°F and for less than 8 hours, record the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s).
- b. By less than 20°F and for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specification 3.0.3 are not applicable.
- c. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by more than 20°F, prepare and submit a Special Report as required by ACTION b. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.14 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limits:

- a. At least once per seven days when the alarm is OPERABLE, and;
- b. At least once per 12 hours when the alarm is inoperable.

TABLE 3.7-6

AREA TEMPERATURE MONITORING

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

TABLE 3.7-6 (Continued)
AREA TEMPERATURE MONITORING

<u>AREA</u>	<u>TEMPERATURE LIMIT (°F)</u>
2. <u>CONTROL BUILDING</u>	
CB-01, Switchgear and Battery Rooms, El 4'6"	≤ 104
CB-02, Cable Spreading Room, El 24'6"	≤ 110
CB-03, Control and Computer Rooms, El 47'6"	≤ 95
CB-04, Chiller Room, El 64'6"	≤ 104
CB-05, Mechanical Equipment Room, El 64-6"	≤ 104
3. <u>CONTAINMENT</u>	
CS-01, Inside Crane Wall, El all except CS-03 and CS-04	≤ 120
CS-02, Outside Crane Wall, El all	≤ 120
CS-03, Pressurizer Cubicle, El all	≤ 130
CS-04, Inside Crane wall, El 51'4" except CS-03 and steam generator enclosures	≤ 120
4. <u>INTAKE STRUCTURE</u>	
CW-01, Entire Building	≤ 110
5. <u>DIESEL GENERATOR BUILDING</u>	
DG-01, Entire Building	≤ 120
6. <u>ESF BUILDING</u>	
ES-01, HVAC and MCC Area, El 36'6"	≤ 110
ES-02, SIH Pump Area, El 21'6"	≤ 110
ES-03, Pipe Tunnel Area, El 4'6"	≤ 110
ES-04, RHS Cubicles, El all	≤ 110
ES-05, RSS Cubicles, El all	≤ 110
ES-06, Motor Driven Auxiliary Feedwater Pump Area, El 24'-6"	≤ 110
ES-07, Turbine Driven Auxiliary Feedwater Pump Area, El 24'6"	≤ 110

TABLE 3.7-6 (Continued)
AREA TEMPERATURE MONITORING

<u>AREA</u>	<u>TEMPERATURE LIMIT (°F)</u>
7. <u>FUEL BUILDING</u>	
FB-02, Fuel Pool Pump Cubicles, El 24'6"	≤ 110
FB-03, General Area, El 52'4"	≤ 104
8. <u>FUEL OIL VAULT</u>	
FV-01, Diesel Fuel Oil Vault	≤ 95
9. <u>HYDROGEN RECOMBINER BUILDING</u>	
HR-01, Recombiner Skid Area, El 24'6"	≤ 125
HR-02, Controls Area, El 24'6"	≤ 110
HR-03, Sampling Area, El 24'6"	≤ 110
HR-04, HVAC Area, El 37'6"	≤ 110
10. <u>MAIN STEAM VALVE BUILDING</u>	
MS-01, Areas Above El 58'0"	≤ 140
MS-02, Areas Below El. 58'0"	≤ 140
11. <u>TURBINE BUILDING</u>	
TB-01, Entire Building	≤ 115
12. <u>TUNNEL</u>	
TN-02, Pipe Tunnel-Auxiliary, Fuel and ESF Building	≤ 112
13. <u>YARD</u>	
YD-01, Yard	≤ 115

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but not less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;
 - c) A flash point equal to or greater than 125°F; and
 - d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.
- 2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79, ASTM-D2622-82 or ASTM-D4294-83.
- f. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;
- g. At least once per 18 months, during shutdown, by:
- 1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;
 - 2) Verifying the generator capability to reject a load of greater than or equal to 595 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 3 Hz;
 - 3) Verifying the generator capability to reject a load of 4986 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection;
 - 4) Simulating a loss-of-offsite power by itself, and:
 - a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and
 - b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 0.8 Hz during this test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 5335 kW;
- 9) Verifying the diesel generator's capability to:
 - a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
 - b) Transfer its loads to the offsite power source, and
 - c) Be restored to its standby status.
- 10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;
- 11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;
- 12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within $\pm 10\%$ of its design interval; and
- 13) Verifying that the following diesel generator lockout features prevent diesel generator starting:
 - a) Engine overspeed,
 - b) Lube oil pressure low (2 of 3 logic),
 - c) Generator differential, and
 - d) Emergency stop.
- h. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 508 rpm in less than or equal to 11 seconds; and
- i. At least once per 10 years by:
 - 1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:
- 1) The parameters in Table 4.8-2a meet the Category B limits,
 - 2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohm, and
 - 3) The average electrolyte temperature of six connected cells is above 60°F.
- c. At least once per 18 months by verifying that:
- 1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,
 - 2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,
 - 3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, and
 - 4) Each battery charger will supply at least the amperage indicated in Table 4.8-2b at 125 volts for at least 24 hours.
- d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;
- e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and
- f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

EMERGENCY CORE COOLING SYSTEMS

BASES

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump and one safety injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps except the required OPERABLE charging pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Surveillance Requirement 4.5.2.C.2 requires that the visual inspection of the containment be performed at least once daily if the containment has been entered that day and when the final containment entry is made. This will reduce the number of unnecessary inspections and also reduce personnel exposure.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 7.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The maximum/minimum solution temperatures for the RWST in MODES 1, 2, 3 and 4 are based on analysis assumptions.

PLANT SYSTEMS

BASES

3/4.7.11 SEALED SOURCE CONTAMINATION (Continued)

plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.14 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of $\pm 2.2^{\circ}\text{F}$.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 100

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated June 2, 1994, as supplemented August 25, 1994, the Northeast Nuclear Energy Company (the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 3 Technical Specifications (TS). The requested changes would revise the Technical Specifications (TS) to remove expired one-time extensions of surveillances, remove an obsolete definition of charging pump operability, and incorporate 11 line item improvements in accordance with the guidance provided in Generic Letter 93-05 (GL 93-05). Several editorial changes have been made to renumber TS pages and delete the blank pages from the TS.

2.0 EVALUATION

2.1 Radiation Monitoring Instrumentation

The licensee determined that the GL 93-05 recommendations regarding radiation monitoring instrumentation are compatible to the Millstone 3 plant. The proposed changes involve granting a relief on the frequency of channel functional tests. These tests do not involve the radiation monitor itself.

In order to decrease the licensee burden and increase the availability of the radiation monitoring systems, the NRC staff finds it acceptable to perform channel functional tests on radiation monitoring equipment on a quarterly basis. The surveillance test intervals for channel checks, source checks, and calibrations would remain the same as in the existing Surveillance Requirements (SRs). This change is consistent with the recommendations in GL 93-05.

2.2 Pressurizer Heaters

The licensee proposed to modify SR 4.4.3.2 by changing the frequency of the pressurizer heater capacity surveillance from at least once per 92 days to at least once each refueling interval. Most pressurizer heaters are in constant

use, both the proportional and to some extent the backup heaters. Therefore, operators should be aware of problems that may arise with pressurizer heaters. In addition, pressurizer heaters are fairly reliable. Measuring the circuit current, from the emergency power supply to the heaters, at least once every 92 days appears to be too frequent. Since the operator is expected to have a grasp of the operational status of the pressurizer heaters, the frequency of once each refueling interval is considered adequate. The surveillance should reveal any heater degradation and it has been shown by operating experience to be acceptable. Therefore, the NRC staff approves of this test being done once each refueling interval.

2.3 Reactor Coolant System (RCS) Isolation Valves

The licensee proposed to modify SR 4.4.6.2.2.b by increasing the time that the unit may remain in cold shutdown before leak testing the RCS isolation valves is required. The time would increase from 72 hours to 7 days.

The existing TS require that RCS isolation valves be tested "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." The SRs are burdensome and can result in occupational exposure even though equipment reliability characteristics suggests longer surveillance test intervals. Longer surveillance test intervals were recommended by the NRC staff in GL 93-05 as a line-item TS improvement to reduce testing during power operation.

The NRC staff believes that making the conduct of this surveillance contingent upon being shut down 72 hours or more has a potential for causing problems resulting from a hurried recovery. It is very likely that a plant could have been operating satisfactorily for more than 9 months without difficulty, and yet must delay restart after a short shutdown period in order to satisfy this SR. In addition, extending the surveillance test interval associated with this surveillance does not significantly alter the associated risk. However, the 7 days will help utilities perform repairs under less stress, but will make an insignificant difference in risk. Therefore, the NRC staff approves of the 72-hour time frame being increased to 7 days for remaining in cold shutdown without testing the RCS isolation valves for leaks.

2.4 High Point Vent Surveillance Testing

The licensee proposed to modify SR 4.4.11.1 by changing the frequency of testing of the RCS vent path block valves from at least once per 92 days to at least once per cold shutdown, if not performed within the previous 92 days. The existing TS requires each RCS vent and block valve to be demonstrated operable every 92 days by operating the valve through one complete cycle of full travel from the control room. Some Westinghouse plants have a problem testing the block valve upstream of the vent valve. When the block valve is opened at pressure, a pressure surge opens the vent valve resulting in a release of reactor coolant. The discharge is routed to either the pressurizer relief tank, a containment ventilation system or directly to the containment atmosphere.

The licensee evaluated the applicability of Catawba TSs Bases (as recommended in GL 93-05) with respect to high point vent surveillance testing, and proposed to revise the frequency of testing the RCS vent path block valves to at least once per cold shutdown, if not performed within the previous 92 days. This change is consistent with the recommendations in GL 93-05, and is acceptable.

2.5 Water Level & Pressure Channel Surveillance Requirements

The licensee examined their channel checks surveillance and operational experience to determine whether there is sufficient basis to justify the extension of the analog channel-operational test frequency.

One action statement states that with one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, the inoperable accumulator must be restored to operable status within 1 hour or the reactor must be in at least hot standby within the next 6 hours and the pressurizer pressure must be reduced to less than 1000 psig within the following 6 hours. This action statement is extremely stringent since only 1 hour is available to correct a problem. Although it does not pose a problem to declare a channel inoperable as a consequence of a channel check (a comparison of the readings from redundant level channels), there is a tendency to require a containment entry to recalibrate the transmitter that indicates low just for the insurance that it provides if the other channel becomes inoperable. This is a consequence of the time limits of the action requirements that require a plant shutdown in 1 hour, which is not enough time to rectify a problem with one instrument. If the allowable outage time were greater with both level instrument channels inoperable, the containment would not be entered just to recalibrate one level transmitter that has experienced excessive drift. A containment entry with only one instrument channel inoperable would result in an unnecessary exposure to radiation.

The combination of redundant level and pressure instrumentation may provide sufficient information so that it may not be worthwhile to always attempt to correct drift associated with one instrument if there were sufficient time to repair one in the event that a second one became inoperable. Because these instruments do not initiate a safety action, it is reasonable to extend the allowable outage time for them. The NRC staff, therefore, finds it acceptable that an additional condition be established for the specific case where "One accumulator is inoperable due to the inoperability of water level and pressure channels," in which the completion time to restore the accumulator to operable status will be 72 hours. While technically inoperable, the accumulator would be available to fulfill its safety function during this time and, thus, this change would have a negligible increase on risk.

The NRC staff has no objection to removing SR 4.5.1.2 from the TS and retaining the specifications in the Millstone 3 Technical Requirements Manual. The accumulator water level and pressure channel analog channel operational test and channel calibration will remain in existing plant procedure requirements.

2.6 Boron Concentration

The existing TSs require that the boron concentration in the accumulators be sampled to verify that it is within its limits after an increase in volume of 1% or more from makeup. The licensee proposed to add a statement to SR 4.5.1.1.b that notes that the surveillance is not required when the volume increase makeup source is the refueling water storage tank (RWST). Normal makeup to an accumulator comes from the RWST which is also borated. No dilution can be caused by adding water from this source as long as the minimum concentration of boron in the RWST is greater than or equal to the minimum boron concentration in the accumulator.

It should not be necessary to verify boron concentration of accumulator inventory after a volume increase of 1% or more if the makeup water is from the RWST and the minimum concentration of boron in the RWST is greater than or equal to the minimum boron concentration in the accumulator, the recent RWST sample was within specifications, and the RWST has not been diluted. The NRC staff finds it acceptable to add clarification to SR 4.5.1.1.b to note when the surveillance is not required.

2.7 Containment Sump

The licensee has proposed to change SR 4.5.2.c.2 to require an inspection of containment at least once daily if the containment has been entered that day, and during the final entry. The existing TSs require 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 suction during loss of coolant conditions. This visual inspection is required to be performed for all areas affected within containment at the completion of each containment entry when containment integrity is established.

The NRC staff recommended in GL 93-05 that containment sump inspections be done at least once daily if the containment has been entered that day and when the final containment entry is made. This is similar to the TSs on air locks for multiple entries. This would reduce the exposure to people doing work in this area by limiting the time spent in the containment. The recommendation would require that a record be kept of all areas visited during the multiple containment entries, but this should not be difficult. This requirement would apply mainly to containment entries when at power, hot standby, or hot shutdown because the requirement states "when containment integrity is established." Therefore, the entries would most likely be made for working on a specific problem in only a few areas. The NRC staff approves the requested change to SR 4.5.2.c.2.

2.8 Containment Spray System

The licensee has proposed to extend the surveillance interval of the air or smoke flow test through each containment spray header from 5 years to 10 years. Each containment spray system must be demonstrated operable at least

once every 5 years in the existing TSs. The test verifies that each spray nozzle is unobstructed.

The system is important to risk, especially for those plants in which the containment spray system performs the dual functions of removing iodine and cooling the containment. Therefore, the heat removal function is redundant. This testing gives no quantitative data on flowrates exiting the nozzles. It only verifies that there is flow, which from the operating data, does not appear to be a problem. An extension from 5 years to 10 years is consistent with the GL 93-05 recommendation. The NRC staff finds it acceptable that this test interval be extended to every 10 years.

2.9 Hydrogen Monitor

The licensee has proposed to modify SR 4.6.4.1 by changing the frequency of the analog channel operational test of each hydrogen monitor and the channel calibration of each hydrogen monitor. The surveillance frequency of calibration of the hydrogen monitors would change from 92 days to each refueling interval, and the analog channel operational test interval would change from 31 days to 92 days.

Hydrogen (or combustible gas) monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions. These monitors are used only after a loss-of-coolant-accident (LOCA) to tell the operator when to initiate the hydrogen recombiners. The hydrogen recombiners are not required for a period of hours to days after a large-break LOCA. The existing TSs require a channel check at least once every 12 hours, an analog channel operational test at least once every 31 days, and a channel calibration at least once every 92 days on a staggered basis. Two gas samples are used - one containing 1% hydrogen, and the other containing 4% hydrogen.

In order to perform the analog channel operation test and calibration, the containment isolation valves which are normally closed must be opened for those plants with the sensor inside the containment. The TSs allow these valves to be "opened on an intermittent basis under administrative control" "Administrative control" means that measures are taken to assure that the control room operators are aware of the position of these valves so that the appropriate action will be taken to close them, if required. There are other operability checks of the system. There are alarms to indicate electronic system and power failures. In addition, operations personnel monitor the indication daily and note any changes in indication. In view of this, the NRC staff recommended in GL 93-05 that the surveillance test interval for hydrogen monitors be extended to a refueling interval, and the analog channel operational test to quarterly.

2.10 Hydrogen Recombiners

The licensee has proposed to modify SR 4.6.4.2.a to change the surveillance test interval for hydrogen recombiner functional tests to once each refueling

interval. The present TSs require testing the hydrogen recombiners at least once every 6 months by performing a functional test. Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination.

The NRC staff finds the frequency acceptable from a reliability standpoint. Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the control room, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

Based on the redundancy and apparent high reliability, the NRC staff approves a surveillance test interval of once each refueling interval.

2.11 Auxiliary Feed Water (AFW) Pumps

The licensee has proposed to modify SR 4.7.1.2.1.a to change the frequency of testing AFW pumps. The current TSs require the monthly testing of AFW pumps. The Boiler and Pressure Vessels Code of the American Society of Mechanical Engineers (ASME Code), Section XI, Paragraph IWP-3400, requires the testing of Class 1, 2, and 3 centrifugal pumps "normally every three months." AFW pumps are the only pressurized water reactor (PWR) pumps required by TSs to be tested more frequently than required by the ASME Code.

The AFW pump takes suction from the condensate storage tank and returns the water to the condensate storage tank through a recirculation line. In most cases, the recirculation line is sized between 5% and 15% of the best estimate point flow. The size is derived from considering the temperature rise in the pump. Pump manufacturers are now recommending that standby pumps be tested at a flow no less than 25% of the best efficiency point flow. This is based on "hydraulic instability" of the pump which is a term for unsteady flow phenomena. Hydraulic instability can cause degradation of the pump and becomes more pronounced as the pump operates further away from its best efficiency point.

It has been suggested in the Electric Power Research Institute Report (NP-4264) and a study done for the NRC's Nuclear Plant Aging Research Program (NUREG/CR-4597), that the size of the recirculation lines be increased. Although a change in recirculation flow from approximately 10% to 25% is the best solution to the problem, reduction in the frequency of testing the AFW pumps from monthly to quarterly could be a reasonable step to reducing the rate of wear. The quarterly testing interval is the frequency specified in the ASME Code, Section XI.

Another consideration is that 42% of AFW pump failures were found during surveillance testing and 29% were found during operation (6% were found during maintenance and, for 23%, the method of detection was not stated). Thus surveillance testing is important in detecting failures in the AFW system. However, surveillance testing also contributes to the problem. The availability of the AFW pump, while related to the conduct of surveillance testing, is not continuously linearly related to surveillance testing. At some point reducing the surveillance test interval will not contribute to an increase in availability. In fact, it could contribute to equipment unavailability. Analyses of AFW pump failures indicates that a monthly surveillance test interval may be contributing to AFW pump unavailability through failures and equipment degradation. The changing of the AFW pump surveillance test interval to quarterly, on a staggered basis, is consistent with the analysis. Conducting the tests on a staggered basis will permit system testing monthly while reducing AFW pump testing to quarterly. The NRC staff believes that a consistent degree of reliability will be maintained with the proposed surveillance interval.

2.12 Administrative Changes

The licensee requested that several administrative changes be made to the TS regarding the removal of blank pages, renumbering pages and sections, and modifying the Index to reflect these changes. Those pages marked "This page intentionally left blank" were deleted which are 3/4 6-16 through 3/4 6-34, 3/4 6-36a, and 3/4 7-30 through 3/4 7-43. Pages 3/4 6-35 through 3/4 6-42 (except for 3/4 6-36a) were renumbered to fill in the spaces where the blank pages were initially. The proposed changes do not affect the configuration, operation, or performance of any system, structure, or component. The administrative changes do not alter the operation of the plant, and are further clarifications of the existing TS requirements.

After discussions with the NRC staff, the licensee identified certain typographical errors and subsequently provided corrected pages. The affected pages are 3/4 3-12, 3/4 3-16, 3/4 5-2, 3/4 6-17, 3/4 7-5, 3/4 7-32, 3/4 7-35, 3/4 7-44, and B3/4 5-2. The content of these pages and the changes made do not affect the initial proposed no significant hazards consideration determination.

2.13 Removal of Cycle Specific Surveillance Extensions

The licensee proposed to change the TS to remove expired one-time extensions of surveillances. These extensions were in effect until September 30, 1993. Since these extensions have expired and the appropriate surveillance performed, the proposed changes do not affect the configuration, operation, or performance of any system, structure or component.

2.14 Clarification of Charging Pump Operability

The licensee proposed to remove an obsolete definition of charging pump operability from the Definition section of the TSs. In doing so, Definition 1.45, "THE CHARGING PUMP OPERABILITY," would be deleted. Subsequently, the

Index would be modified to reflect this change. The definition was added to the TSs for Cycle 4 of operation and when the licensee relied on a temporary heat source. The licensee has completed the necessary modifications, and therefore, no longer needs this definition. This change is administrative and does not alter the operation of the plant, and is a further clarification of the existing TS requirements.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 47170). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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