Docket file

June 8, 1993

Docket No. 50-423

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

Dear Mr. Opeka:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M86148)

The Commission has issued the enclosed Amendment No. 79 to Facility Operating License No. NPF-49 for Millstone Nuclear Power Station, Unit No. 3, in response to your application dated March 30, 1993, as supplemented April 20 and 27, 1993.

The amendment revises the Technical Specifications to extend the interval for surveillance testing of 42 instrumentation and control items presently required to be tested by June 13, 1993, or later, until the next refueling outage, but no later than September 30, 1993.

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

Enclosures:
1. Amendment No. 79 to NPF-49
2. Safety Evaluation

cc w/enclosures:
See next page

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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. 79 , 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

John F. Stolz, 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: June 8, 1993

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

June 8, 1993

Docket No. 50-423

Mr. John F. Opeka
Executive Vice President, Nuclear
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Sincerely,

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

Enclosures:
1. Amendment No. 79 to NPF-49
2. Safety Evaluation

cc w/enclosures:
See next page

AMENDMENT NO. 79

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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. 79
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 March 30, 1993, as supplemented April 20 and 27, 1993, 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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PDR ADOCK 05000423
P PDR

ATTACHMENT TO LICENSE AMENDMENT NO. 79

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	Insert
3/4 2-20	3/4 2-20
3/4 2-23	3/4 2-23
3/4 3-1	3/4 3-1
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-14	3/4 3-14
3/4 3-16	3/4 3-16
3/4 3-36	3/4 3-36
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38
3/4 3-39	3/4 3-39
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-49	3/4 3-49
3/4 3-53	3/4 3-53
3/4 3-58	3/4 3-58
3/4 3-62	3/4 3-62
3/4 3-63	3/4 3-63
3/4 3-68	3/4 3-68
3/4 4-21	3/4 4-21
3/4 4-39	3/4 4-39
3/4 4-43	3/4 4-43
3/4 4-45	3/4 4-45
3/4 5-4	3/4 5-4
3/4 8-7	3/4 8-7
3/4 8-12	3/4 8-12

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months.* Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel (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.

*Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14, 20)	N.A.	1, 2, 3*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5, 20)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4, 20)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4, 20)	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4, 20)	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 (20)	Q(18)	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R (20)	Q(18)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R (20)	Q	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	S	R (20)	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	S	R(20)	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, 20)	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, 20)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	R(4, 20)	R	N.A.	N.A.	1
e. Power Range Neutron Flux, P-10	N.A.	R(4, 20)	R	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
20. Three Loop Operation Bypass Circuitry	N.A.	N.A.	N.A.	R(20)	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).

(20) Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

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.

*Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

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

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

SURVEILLANCE REQUIREMENTS

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

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

*Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

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

2. A nominal 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

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

*Except that the surveillance requirement due no later than June 13, 1993 may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R(4)	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(4)	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R(4)	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(4)	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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R(4)	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.(4)							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R(4)	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(4)	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
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(4)	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R(4)	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(4)	Q	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Safety Injection Actuation Logic	N.A.	N.A.	N.A.	R(4)	N.A.	N.A.	N.A.	1, 2
d. T_{ave} Low Coincident with Reactor Trip (P-4)	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R(4)	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(4)	N.A.	N.A.	N.A.	All
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	R(4)	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

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Control Building Isolation (Continued)								
e. Control Building Inlet Ventilation Radiation	S	R	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(4)	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(4)	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

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.

(4) Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

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

TABLE NOTATIONS

* With fuel in the fuel storage pool area.

** Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

* Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

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.*

* Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

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

** Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

**Electronic calibration from the ICC cabinets only.

***Except that the surveillance requirements due no later June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

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.*

* Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

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 Particular Radioactivity Monitoring System is OPERABLE; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the followng 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.*

* Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months**; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

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

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

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

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

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* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

** Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of two vent valves(s) and one block valve powered from emergency busses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months* by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

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

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

* Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

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

b. At least once per 31 days by:

1) Verifying that the ECCS piping, except for the RSS pump, heat exchanger and associated piping, is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

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

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

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

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

d. At least once per 18 months* by:

1) Verifying automatic 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.

* Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 5335 kW;

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10) Verifying* that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

12) Verifying* that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval; and

13) Verifying that the following diesel generator lockout features prevent diesel generator starting:

   a) Engine overspeed,

   b) Lube oil pressure low (2 of 3 logic),

   c) Generator differential, and

   d) Emergency stop.

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

* Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier.

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.

* Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage but no later than September 30, 1993, whichever is earlier.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 79

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 March 30, 1993, as supplemented April 20 and April 27, 1993, the Northeast Nuclear Energy Company (NNECO/the licensee) proposed amending the operating license of the Millstone Nuclear Power Station, Unit No. 3. The current Technical Specifications (TS) for Millstone Unit 3, specify 18-month surveillance requirements for a number of instrumentation functions in the reactor control system (RCS), engineered safety features actuation system (ESFAS), radiation monitoring system, seismic monitoring system, remote shutdown monitoring system, loose-part detection system, reactor coolant system, emergency core cooling systems, and electrical power system. To comply with the requirements of the current TS, 18-month surveillance tests on these system instrumentation channels need to be performed between June 13, 1993, and September 23, 1993. Originally, the Millstone, Unit 3, refueling outage was scheduled to start in November 1992. However, because of an unusually long maintenance outage on the service water system and erosion/corrosion work during 1991, and two limited outages in 1992, NNECO rescheduled the refueling outage to begin July 31, 1993, and requested that the 18-month surveillance requirements be extended by adding the following note to the TS to avoid plant shutdown:

> "Except that the surveillance requirement due no later than June 13, 1993, may be deferred until the next refueling outage, but no later than September 30, 1993, whichever is earlier."

The proposed amendment would permit a one-time extension of the 18-month surveillance requirements by a minimum of 7 days and a maximum of 109 days including the 25 percent allowance for the 18-month period permitted by the current TS.

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2.0 EVALUATION

In their evaluation of the proposed TS amendment, NNECO provided a combination of the following justifications:

A. Probabilistic Risk Assessment (PRA).

The staff has reviewed the Individual Plant Examination of the Millstone Nuclear Power Station, Unit No. 3, and has issued a safety evaluation report on May 5, 1992. In the evaluation, the staff concluded that the internal events portion of the individual plant examination process is consistent with the intent of Generic Letter 88-20. No specific unresolved safety issue or generic safety issue was proposed for resolution as part of the individual plant examination of the Millstone Nuclear Power Station, Unit No. 3.

From the present evaluation of the licensee's analysis of the proposed one-time extension of the surveillance requirements, the staff has determined that the licensee's method of analysis is appropriate, the data used are relevant, and the assumptions are conservative. The staff further determined that the licensee's estimated increases in the total core damage frequency, as a result of the proposed extension of the surveillance requirements for the three interlocks under consideration, are negligibly small.

On the basis of the above evaluation, the staff concludes that the licensee's probabilistic safety assessment as supporting justification for the proposed one-time extension of the surveillance test intervals for reactor trip system interlocks, units 17C and 17D, and engineered safety features actuation system interlock, unit 9C, is acceptable.

In addition, for reactor trip system interlock units 17C and 17D as well as for safety feature actuation system interlock, unit 9C, the licensee has examined the history of completed surveillance tests. In none of these cases were any failures detected. Therefore, for reactor trip system interlock units 17C and 17D, and for engineered safety system interlock, unit 9C, the proposed extension of the surveillance interval is supported by the probabilistic safety assessment and by the history of successfully completed surveillance tests.

B. Instrument Loops using sensors in the Rosemount Program.

NNECO uses an on-line monitoring program that detects "loss of fill-oil" for each Rosemount Model 1153 and 1154 transmitter and for several Veritrack transmitters. Such on-line monitoring substantially reduces undetected failures of the transmitters.

C. Historical Data.

For instrument loops using sensors not in the Rosemount Program, NNECO used previous surveillance data to establish the worst case drift. The data was then extrapolated to the end of the proposed surveillance interval extension, and compared to the allowable drift for the sensor.

D. Analog Channel Operational Test and Channel Checks.

NNECO took credit for the current analog channel operational tests and channel checks which provide confidence with regard to the functionality of the process loop including the sensors.

E. Manual Initiation.

NNECO took credit for the manual trip instrumentation that was not susceptible to drift and for which no credit was taken in the accident analysis. This instrumentation includes manual initiation of safety injection, containment spray, containment isolation, steam line isolation, and control building isolation.

The above justification discussion is similar to others approved by the staff for similar TS instrumentation surveillance extension requests. The overall justification provides a defense-in-depth argument for ensuring plant safety over the limited period of operation beyond the date currently required for the surveillances.

The NRC staff has reviewed the above justifications and concur with the licensee that they provide reasonable assurance that the necessary instrumentation functions can be accomplished during the short period of time beyond the currently required surveillance date. The proposed TS change is, therefore, acceptable.

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 surveillance requirement. 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 (58 FR 26988). 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.

Principal Contributors: S. Mazumdar and J. Wing

Date: June 8, 1993