MAR 23 1977

Docket No. 50-336

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Northeast Nuclear Energy Company
ATTN: Mr. D. C. Switzer
President
P. O. Box 270
Hartford, Connecticut 06101

Gentlemen:

The Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your applications dated November 29, 1976, two applications dated December 6, 1976 and December 13, 1976.

The amendment will provide (1) allowance for the inoperability of two containment air recirculation and cooling units, (2) a change to the action required in the event that the core average burnup exceeds 500 effective full power days, (3) deletion of the response times associated with Manual Initiation of Engineered Safety Features, (4) deletion of the minimum duration for containment leakage supplemental tests, and (5) correction of various spelling and editorial errors in the Technical Specifications.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 25
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

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OFFICE➤		ORB#3	ORB#3	OELD	STS&G/DOR	ORB#3
SURNAME➤		CParrish CP	DJaffe:acr	J. R. GRAY	JMcGough	GLear G
DATE➤		3/ 1 /77	3/ 8 /77	3/ 15 /77	3/ 8 /77	3/ 22 /77

NRC FORM 318 (9-76) NRCM 0240

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Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 25
2. Safety Evaluation
3. FEDERAL REGISTER Notice

cc w/encls:
See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE➤	ORB #3	ORB #3	OELD	STSG/DOR	ORB #3	
SURNAME➤	*CParrish	*DJaffe:acr	*	*McGough	GLear	
DATE➤	3/7/77	3/8/77	3/15/77	3/8/77	3/ /77	

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THE CONNECTICUT LIGHT AND POWER COMPANY,
THE HARTFORD ELECTRIC LIGHT COMPANY,
WESTERN MASSACHUSETTS ELECTRIC COMPANY, AND
NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. DPR-65

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

   A. The applications for amendment by The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company (the licensees), dated November 29, 1976, November 30, 1976, two applications dated December 6, 1976, and December 13, 1976, comply 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 applications, 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.

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. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 25 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 23, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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 area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 2-11
3/4 3-21
3/4 3-28
3/4 3-30
3/4 4-9
3/4 6-1
3/4 6-3
3/4 6-8
3/4 6-14
3/4 6-16
3/4 6-26
3/4 6-28
3/4 7-17
3/4 10-1
3/4 10-3
B 3/4 3-2
B 3/4 6-2
B 3/4 6-3

FUEL RESIDENCE TIME

LIMITING CONDITION FOR OPERATION

3.2.4 The core average fuel burnup shall be limited to $\leq$ 500 Effective Full Power Days during the initial fuel cycle.

APPLICABILITY: MODE 1.

ACTION:

With the core average fuel burnup determined to exceed 500 Effective Full Power Days, be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 The core average fuel burnup, based on gross thermal energy generation, shall be determined by calculation at least once per 31 days.

DNB MARGIN

LIMITING CONDITION FOR OPERATION

3.2.5 The DNB margin shall be preserved by maintaining the cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX within the limits specified in Table 3.2-1 and Figure 3.2-4.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its specified limits, restore the parameter to within its above specified limits within 2 hours or reduce THERMAL POWER to $\leq$ 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX shall be determined to be within the limits of Table 3.2-1 and Figure 3.2-4 at least once per 12 hours.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Isolation	Not Applicable
Enclosure Building Filtration System	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. SRAS	
Containment Sump Recirculation	Not Applicable
e. EBFAS	
Enclosure Building Filtration System	Not Applicable
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0*/30.0**
b. Containment Isolation	$\leq$ 7.5
c. Enclosure Building Filtration System	$\leq$ 35.0*/35.0**
3. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 30.0*/30.0**
b. Containment Isolation	$\leq$ 7.5
c. Enclosure Building Filtration System	$\leq$ 35.0*/35.0**

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 34.5*/20.5**
5. Containment Radiation-High	
a. Containment Purge Valves Isolation	$\leq$ Counting period plus 7.5
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	$\leq$ 6.9
b. Feedwater Isolation	$\leq$ 60
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	$\leq$ 120

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

**Diesel generator starting and sequence loading delays not included. Offsite power available.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Storage Criticality Monitor	1	*	≤ 2 x background	10^{-1} - 10^{+4} mR/hr	13
Ventilation System Isolation	2	#	≤ 2 x background	10^{-1} - 10^{+4} mR/hr	15
2. PROCESS MONITORS					
a. Containment Atmosphere-Particulate	1	ALL MODES	≤ 36,000 cpm/hr	10 - 10^{+6} cpm/hr	14 and (a)
b. Containment Atmosphere-Gaseous	1	ALL MODES	≤ 38,000 cpm** ≤ 380 cpm***	10 - 10^{+6} cpm	14 and (a)
c. Spent Fuel Storage-Particulate	1	*	≤ 13,000 cpm	10 - 10^{+6} cpm	14
d. Spent Fuel Storage-Gaseous	1	*	≤ 835 cpm	10 - 10^{+6} cpm	14

*With fuel in storage building.
**With personnel not in containment.
***With personnel in containment.
#With irradiated fuel in the storage pool.

TABLE 3.3-6 (Continued)

TABLE NOTATION

(a) - During MODE 6, also comply with the ACTION requirements of Specification 3.9.9, as applicable.

ACTION 13 - With the number of area monitors OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 14 - With the number of process monitors OPERABLE less than required by the Minimum Channels OPERABLE requirement either (a) obtain and analyze grab samples of the monitored parameter at least once per 24 hours, or (b) use a Constant Air Monitor to monitor the parameter.

ACTION 15 - With the number of area monitors OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.13.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Storage				
Criticality Monitor	S	R	M	*
Ventilation System Isolation	S	R	M	#
2. PROCESS MONITORS				
a. Containment Atmosphere-Particulate	S	R	M	ALL MODES
b. Containment Atmosphere-Gaseous	S	R	M	ALL MODES
c. Spent Fuel Storage-Particulate	S	R	M	*
d. Spent Fuel Storage-Gaseous	S	R	M	*

*With fuel in storage building;

#With irradiated fuel in the storage pool.

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of three OPERABLE rhodium detectors.

APPLICABILITY: When the incore detection system is used for:

a. Recalibration of the excore axial flux offset detection system,

b. Monitoring the AZIMUTHAL POWER TILT,

c. Calibration of the power level neutron flux channels, or

d. Monitoring the linear heat rate.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Recalibration of the excore axial flux offset detection system,

2. Monitoring the linear heat rate pursuant to specification 4.2.1.3,

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through steam generators, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the containment atmosphere particulate radioactivity at least once per 12 hours.

b. Monitoring the containment sump inventory at least once per 12 hours,

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

REACTOR COOLANT SYSTEM

CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: ALL MODES.

ACTION:

MODES 1, 2, 3 and 4

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in COLD SHUTDOWN within 36 hours.

MODES 5 and 6

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

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of these parameters at the frequencies specified in Table 4.4-1.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated at least once per 31 days by verifying that:

a. All penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.1,

b. The equipment hatch is closed and sealed, and

c. The containment air lock is OPERABLE per Specification 3.6.1.3.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.50 percent by weight of the containment air per 24 hours at P_a, 54 psig.

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

c. A combined leakage rate of $\leq 0.017\ L_a$ for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, or (c) with the combined bypass leakage rate exceeding $0.017\ L_a$, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

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

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at P_a (54 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet .75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet .75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .75 L_a at which time the above test schedule may be resumed.

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

1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a.

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

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a (54 psig).

d. Type B and C tests shall be conducted at P_a (54 psig) at intervals no greater than 24 months except for tests involving air locks.

e. The combined bypass leakage rate shall be determined to be < 0.017 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (54 psig) during each Type A test.

f. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

SURVEILLANCE REQUIREMENTS (Continued)

g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to determine the inaccuracy of the measured leakage rates due to maximum measurement accuracy and instrument repeatability; the measured leakage rates shall be adjusted to include the measurement error.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 6 months by conducting an overall air lock leakage test at P_a (54 psig) and by verifying that the overall air lock leakage rate is within its limit, and

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -12 inches Water Gauge and +2.1 PSIG.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure in excess of or below the limits above, restore the internal pressure to within the limits within 1 hour or be in HOT STANDBY within the next 4 hours; go to COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump operates for at least 15 minutes,

4. Cycling each testable, automatically operated valve in each spray system flow path through at least one complete cycle,

5. Verifying that upon a sump recirculation actuation signal the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established, and

6. Verifying that all accessible manual valves not locked, sealed or otherwise secured in position and all remote or automatically operated valves in each spray system flow path are positioned to take suction from the RWST on a Containmenet Pressure--High-High signal.

b. At least once per 18 months, during shutdown, by cycling each power operated valve in the spray system flow path not testable during plant operation through at least one complete cycle of full travel.

c. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT AIR RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Four containment air recirculation and cooling units shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one containment air recirculation and cooling unit inoperable and both containment spray systems OPERABLE, restore the inoperable air recirculation and cooling unit to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

b. With one containment air recirculation and cooling unit inoperable and one containment spray system inoperable, restore either the inoperable air recirculation and cooling unit or the inoperable spray system to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

c. With two containment air recirculation and cooling units inoperable and both containment spray systems OPERABLE, restore at least one of the inoperable air recirculation and cooling units to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each containment air recirculation and cooling unit shall be demonstrated OPERABLE at least once per 31 days on a STAGGERED TEST BASIS by:

a. Starting, in low speed, each unit from the control room,

b. Verifying that each unit operates for at least 15 minutes, and

c. Verifying a cooling water flow rate of $\geq$ 500 gpm to each cooling unit.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves specified in Table 3.6-2 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-2 inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate the affected penetration(s) within 4 hours by use of a deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of a closed manual valve(s) or blind flange(s); or

d. Be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 The isolation valves specified in Table 3.6-2 as testable during plant operation shall be demonstrated OPERABLE:

a. At least once per 92 days by:

1. Exercising each power operated valve through one complete cycle of full travel and measuring the isolation time, and

2. Exercising each manual valve, except those that are closed, through one complete cycle of full travel.

b. Immediately prior to returning the valve to service after maintenance, repair or replacement work is performed on the

SURVEILLANCE REQUIREMENTS (Continued)

valve or its associated actuator, control or power circuit by performance of the applicable cycling test, above.

4.6.3.1.2 Each isolation valve specified in Table 3.6-2 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position,

b. Verifying that on a Containment Radiation-High signal, all containment purge valves actuate to their isolation position,

c. Exercising each power operated valve not testable during plant operation, through one complete cycle of full travel and measuring its isolation time, and

d. Exercising each manual valve not locked, sealed or otherwise secured in position through at least one complete cycle of full travel.

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

ENCLOSURE BUILDING FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two separate and independent enclosure building filtration systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one enclosure building filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each enclosure building filtration 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 filter and charcoal adsorber train and verifying that the train operates for at least 10 hours with the heaters on.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 12 months or after every 720 hours of system operation and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following painting, fire or chemcial release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 9000 cfm $\pm$ 10%.

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 9000 cfm $\pm$ 10%.

3. Subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying removal efficiency of $\geq$ 90% for radioactive methyl iodide at an air flow velocity of 0.67 ft/sec $\pm$ 20% with an inlet methyl iodide concentration of 0.05 to 0.15 mg/m^3, $\geq$ 95% relative humidity, and $\geq$ 125°F; other test conditions shall be in accordance with USAEC RDT Standard M-16-1T, June 1972. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 9000 cfm $\pm$ 10% during system operation.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is $\leq$ 6 inches Water Gauge while operating the ventilation system at a flow rate of 9000 cfm $\pm$ 10%.

2. Verifying that the air flow distribution to each HEPA filter and charcoal adsorber is within $\pm$ 20% of the averaged flow per unit.

3. Verifying that the filtration system starts automatically on a Enclosure Building Filtration Actuation Signal (EBFAS).

4. Verifying that each system produces a negative pressure of $\geq$ 0.25 inches W. G. in the Enclosure Building Filtration Region within 1 minute after an Enclosure Building Filtration Actuation Signal (EBFAS).

ENCLOSURE BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.2 ENCLOSURE BUILDING INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without ENCLOSURE BUILDING INTEGRITY, restore ENCLOSURE BUILDING INTEGRITY within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 ENCLOSURE BUILDING INTEGRITY shall 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, then at least one door shall be closed.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 12 months or after every 720 hours of system operation and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housing, or (3) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm $\pm$ 10%.

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm $\pm$ 10%.

3. Subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying a removal efficiency of $\geq$ 90% for radioactive methyl iodide at an air flow velocity of 0.67 ft/sec $\pm$ 20% with an inlet methyl iodide concentration of 0.05 to 0.15 mg/m^3, $\geq$ 95% relative humidity, and $\geq$ 125°F; other test conditions shall be in accordance with USAEC RDT Standard M-16-1T, June 1972. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 2000 cfm $\pm$ 10% during system operation.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the ventilation system at a flow rate of 2000 cfm $\pm$ 10%.

2. Verifying that on a recirculation signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided:

a. Reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA's, and

b. All part length CEAs are at least 90% withdrawn and OPERABLE.

APPLICABILITY: MODES 2 and 3.

ACTION:

a. With the reactor critical ($K_{eff} \geq 1.0$) and with less than the above reactivity equivalent available for trip insertion or the part length CEAs not within their withdrawal limits, immediately initiate and continue boration at $\geq$ 44 gpm until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With the reactor subcritical ($K_{eff} < 1.0$) by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 44 gpm until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated OPERABLE by verifying its CEA drop time to be $\leq$ 3.0 seconds within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 The part length CEAs shall be demonstrated OPERABLE by moving each part length CEA $\geq$ 10 steps within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT AND INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The requirements of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2 and 3.2.3 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1, being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2 and 3.2.3. are suspended, immediately:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1 or

b. Be in HOT STANDBY within 2 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2 or 3.2.3 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THEMAL POWER in which the requirements of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2 or 3.2.3 are suspended.

SPECIAL TEST EXCEPTIONS

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5 percent of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE power range neutron flux monitoring channels are set at $\leq$ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation shown on Figure 3.4-2.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the unacceptable region of operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit; perform the analysis required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The Reactor Coolant System shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.

4.10.3.2 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour.

4.10.3.3 Each wide range logarithmic and power level channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at ≤ 20% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each wide range logarithmic and power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination therof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served

RADIATION MONITORING INSTRUMENTATION (Continued)

by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs."

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 54 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and 3.6.1.2. The limitations on the air locks allow entry and exit into and out of the containment during operation and ensure through the surveillance testing that air lock leakage will not become excessive through continuous usage.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that the containment peak pressure does not exceed the design pressure of 54 psig during LOCA conditions.

The maximum peak pressure obtained from a LOCA event is 51.2 psig. The limit of 2.1 psig for initial positive containment pressure will limit the total pressure to less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment air temperature ensures that the containment peak air temperature does not exceed the design temperature of 288°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 51.2 psig in the event of a LOCA. The measurement of containment tendon lift off force, the visual and metallurgical examination of tendons, anchorages and liner and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's strucutral integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Strucutres".

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.2.2 CONTAINMENT AIR RECIRCULATION SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjuntion with the containment spray systems during post-LOCA conditions.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

BASES

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the purge system is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA."

The post-incident recirculation systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

Introduction

By applications for license amendments dated November 29, 1976, November 30, 1976, two applications dated December 6, 1976, and December 13, 1976, Northeast Nuclear Energy Company (NNECO) requested changes to the Technical Specifications for Millstone Unit No. 2. The requested changes provide for (1) allowance for the inoperability of two containment air recirculation and cooling units, (2) a change to the action required in the event that the core average burnup exceeds 500 effective full power days, (3) deletion of the response times associated with Manual Initiation of Engineered Safety Features, (4) deletion of the minimum duration for containment leakage supplemental tests, and (5) correction of various typographical and editorial errors in the Technical Specifications.

In the course of reviewing the November 29, 1976 application, we found it necessary to make changes in the proposed Technical Specifications. NNECO has reviewed and approved the changes to the proposed Technical Specifications.

Discussion and Evaluation

Our discussion and evaluation of NNECO's proposed changes to the Technical Specifications is contained in the following sections:

1. Allowance for the Inoperability of Two Containment Air Recirculation and Cooling Units

   By application dated November 29, 1976, NNECO has requested a change to Technical Specification 3.6.2.2 to allow two air recirculation and cooling units to be inoperable for 48 hours. The post-loss-of-coolant-accident (LOCA) containment cooling capability for Millstone Unit No. 2 consists of (1) two containment spray systems each with a 50% containment cooling capacity and (2) four air recirculation and cooling units each with a 33% containment cooling capacity. An analysis has been performed of the effect on post-LOCA containment cooling capability of various allowable combinations of inoperable containment spray and air recirculation and cooling units. The limiting equipment configuration permitted is addressed in Technical Specification 3.6.2.2.b which allows one containment spray and one air recirculation and cooling unit to be inoperable for up to 48 hours. In the unlikely event of a LOCA during this 48 hour period, with the worst single failure, the remaining containment cooling capability consists of one containment air recirculation and cooling unit which provides a 33% capability for containment cooling. This results from the assumption that the worst single failure involves the diesel generator which powers a core spray and two air recirculation and cooling units. The remaining diesel generator would power a single air recirculation and cooling unit with 33% containment cooling capability.

NNECO's request to allow two containment air recirculation and cooling units to be inoperable for 48 hours, in addition to the existing combinations of allowable containment spray and air recirculation and cooling unit inoperability, would provide a 50% capability for post-LOCA containment cooling assuming the worst single failure. NNECO's request is therefore more conservative than the most limiting equipment configuration permitted by existing Technical Specifications in that it provides 50% cooling capacity as compared with 33% capacity for the most limiting equipment configuration.

Accordingly, we find that the proposed change does not increase the probability or consequences of accidents previously considered nor does it decrease the safety margin with regard to containment cooling capability. The proposed change to Technical Specification 3.6.2.2, which allows two containment air recirculation and cooling units to be inoperable for 48 hours is therefore acceptable.

2. <u>Required Action in the Event that the Core Average Fuel Burnup Exceeds 500 Effective Full Power Days</u>

Technical Specification 3.2.4 requires that the reactor be in cold shutdown within 36 hours in the event that the core average burnup exceeds 500 effective full power days. By application dated November 30, 1976, NNECO has requested that the reactor be placed in hot standby, instead of cold shutdown, in the event that the core average burnup exceeds 500 effective full power days.

The Bases provided for Technical Specification 3.2.4 states that, "The limitation on fuel burnup during the initial fuel cycle insures that fuel cladding collapse will not occur. Performance data from similar fuel rods and analyses of the installed fuel rods show that cladding collapse will not occur until well beyond the proposed first cycle of operation which is about 11,200-12,000 Effective Full Power Hours. However, operation beyond the first cycle will require further analyses."

Thus, the intent of Technical Specification 3.2.4 is to limit the burnup of the Millstone Unit No. 2 first cycle to 12,000 effective full power hours (500 effective full power days) to assure that no fuel clad collapse will occur. Placing the reactor in hot standby (subcritical with an average coolant temperature greater than 300°F) is as effective as placing the reactor in cold shutdown (subcritical with an average coolant temperature less than 200°F) in terms of preventing further core burnup; in both cases the reactor is subcritical so that further burnup cannot occur.

We find that NNECO's proposal, to change the required action in the event that a core average burnup of 500 effective full power days is exceeded, is acceptable since the probability of clad collapse is not affected by the required remedial action that must be taken in the event that a burnup of 500 effective full power days is exceeded.

3. <u>Deletion of the Response Times Associated with Manual Initiation of Engineered Safety Features</u>

In an application dated December 6, 1976, NNECO requested that the response times associated with Manual Initiation of Engineered Safety Features be deleted from Table 3.3-5 of the Technical Specifications. The actions for which the manual response times would be deleted are ECCS Safety Injection, Containment Isolation, Enclosure Building Filtration System, Containment Spray, and Containment Sump Recirculation.

The Technical Specification Bases state that the measurement of response times provides assurance that each protective and Engineered Safety Feature action is completed within the time limit assumed in the accident analyses. However, as noted by NNECO in the December 6, 1976 application, all of the above actions are initiated automatically and are treated as such in the Safety Analyses. Therefore, it is not appropriate to list or require confirmation via testing of manual response times for these actions. Response times for each of the above actions based upon automatic initiation is included in Table 3.3-5 and we conclude that this is adequate.

Based on our review, we conclude that the proposed deletion of response times associated with Manual Initiation of Engineered Safety Features is acceptable.

4. Deletion of the Minimum Duration for Containment Leakage Supplemental Tests

By application dated December 6, 1976, NNECO requested that the minimum duration of the Containment Leakage Supplemental Test, specified in Technical Specification 4.6.1.2.c.2. be modified.

The purpose of the Containment Leakage Supplemental Test is to provide a method to confirm the calculations used to determine the containment leakage rate following completion of the Containment Leakage (Type A) Test. NNECO has found that the Supplemental Containment Leakage Test can be conducted in a considerably shorter time than the minimum duration of 6 hours plus the stabilization period required by Technical Specification 4.6.1.2.c.2. This results in needlessly extending the time required to conduct the test. Moreover, 10 CFR Part 50, Appendix J, Section A.3.b., requires only that, "The supplemental test method selected shall be conducted for sufficient duration to establish accurately the change in leakage rate between the Type A and supplemental test". NNECO proposes to substantially adopt the wording of Section A.3.b. in Technical Specification 4.6.1.2.c.2.

We conclude from the above that deletion of minimum duration for the supplemental containment leakage test will not affect the conduct of the test. Moreover, adopting the wording from Section A.3.b. is appropriate since the existing Technical Specification Bases state that, "The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix 'J' of 10 CFR 50". Accordingly, the change to Technical Specification 4.6.1.2.c.2., to delete the minimum duration for Supplemental Containment Leakage tests and substitute wording consistent with Appendix J to 10 CFR Part 50, is acceptable.

5. Correction of Spelling and Editorial Errors

By application dated December 13, 1976, NNECO requested a change to the Technical Specifications which would correct a number of typographical and editorial errors. The proposed changes have no affect on reactor safety and are thus acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to

10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 23, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

NORTHEAST NUCLEAR ENERGY COMPANY,
THE CONNECTICUT LIGHT AND POWER COMPANY,
THE HARTFORD ELECTRIC LIGHT COMPANY, AND
WESTERN MASSACHUSETTS ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 25 to Facility Operating License No. DPR-65 issued to Northeast Nuclear Energy Company, The Connecticut Light and Power Company, The Hartford Electric Light Company, and Western Massachusetts Electric Company, which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2, located in the Town of Waterford, Connecticut. The amendment is effectve as of the date of issuance.

The amendment will provide (1) allowance for the inoperability of two containment air recirculation and cooling units, (2) a change to the action required in the event that the core average burnup exceeds 500 effective full power days, (3) deletion of the response times associated with Manual Initiation of Engineered Safety Features, (4) deletion of the minimum duration for containment leakage supplemental tests, and (5) correction of various typographical and editorial errors in the Technical Specifications.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or a negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated November 29, 1976, November 30, 1976, two applications dated December 6, 1976, and December 13, 1976, (2) Amendment No. 25 to License No. DPR-65, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Waterford Public Library, Rope Ferry Road, Waterford, Connecticut 06385.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 23 day of March 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors