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

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. DPR-65

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 December 21, 1994, as supplemented February 22, 1995, 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.

2. Accordingly, Facility Operating License No. DPR-65 is hereby amended by modifying paragraph 2.C.(3) on page 4 to read as follows:

   (3) Fire Protection

   The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report and as approved in the SER dated September 19, 1978, and supplements dated October 21, 1980, November 11, 1981, October 31, 1985, April 15, 1986, January 15, 1987, April 29, 1988, July 17, 1990, and November 3, 1995, subject to the following provisions.

   The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

   The license is also 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 Appendix A, as revised through Amendment No. 191 , 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 issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip F. McKee

Phillip F. McKee, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: 1. Page 4 of License
2. Changes to the Technical Specifications

Date of Issuance:

*Page 4 is attached, for convenience, for the composite license to reflect this change.

(3) Fire Protection

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report and as approved in the SER dated September 19, 1978, and supplements dated October 21, 1980, November 11, 1981, October 31, 1985, April 15, 1986, January 15, 1987, April 29, 1988, July 17, 1990, and November 3, 1995, subject to the following provisions.

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Millstone Nuclear Power Station Physical Security Plan," with revisions submitted through March 29, 1988; "Millstone Nuclear Power Station Suitability, Training and Qualification Plan," with revision submitted through July 21, 1986; and "Millstone Nuclear Power Station Safeguards Contingency Plan," with revisions submitted through October 30, 1985. Changes made in accordance with 10 CFR 73.55 will be implemented in accordance with the schedule set forth therein.

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

Remove	Insert
V	V
VIII	VIII
XIII	XIII
3/4 3-43	3/4 3-43
3/4 3-44	3/4 3-44
3/4 3-45	3/4 4-45
3/4 3-45a	3/4 3-45a
3/4 7-33	3/4 7-33
3/4 7-34	3/4 7-34
B 3/4 3-4	B 3/4 3-4
B 3/4 7-6	B 3/4 7-6
B 3/4 7-7	B 3/4 7-7
6-2	6-2
6-4	6-4
6-5	6-5
6-6	6-6
6-8	6-8
6-20	6-20

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.7.9 Deleted

3/4.7.10 Deleted

3/4.7.11 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.11 The ultimate heat sink shall be OPERABLE with an average water temperature of less than or equal to 75°F at the Unit 2 intake structure.

APPLICABILITY: MODES 1, 2, 3, AND 4

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.11 The ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the average water temperature at the Unit 2 intake structure to be within limits.

b. At least once per 6 hours by verifying the average water temperature at the Unit 2 intake structure to be within limits when the average water temperature exceeds 70°F.

3/4.3.3.6 DELETED

3/4.3.3.7 DELETED

3/4.3.3.8 Accident Monitoring Instrumentation

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. Due to the size and location of the steam generator hydraulic snubbers, regular removal and testing as specified for hydraulic and mechanical snubbers would represent a significant undertaking during each refueling outage. As such, these snubbers have been treated separately and are tested and refurbished as a group in accordance with the manufacturer's recommended preventative maintenance program.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber.

The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber reliability, a representative sample of the installed snubbers will be tested during plant shutdowns at eighteen (18) month intervals. Observed failures of these sample snubbers shall require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc....). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.9 DELETED

3/4.7.10 DELETED

3/4.7.11 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink temperature ensure that sufficient cooling capacity is available to either,

1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on maximum temperature are based on a 30-day cooling water supply to safety related equipment without exceeding their design basis temperature.

FACILITY STAFF (CONTINUED)

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. (Table 6.2-1)

e. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions. These procedures should follow the general guidance of the NRC Policy Statement on working hours (Generic Letter No. 82-12).

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for:

a. If the Operations Manager does not hold a senior reactor operator license for Millstone Unit No. 2, then the Operations Manager shall have held a senior reactor operator license at a Pressurized Water Reactor and an individual serving in the capacity of the Assistant Operations Manager shall hold a senior reactor operator license for Millstone Unit No. 2.

b. The Shift Technical Advisor (STA) who shall meet the requirements of Specification 6.3.1.b.1 or 6.3.1.b.2.

1. Dual-role individual: Must hold a senior reactor operator's license at Millstone Unit No. 2, meet the STA training criteria of NUREG-0737, Item I.A.1.1, and meet one of the following educational alternatives:

a. Bachelor's degree in engineering from an accredited institution;

b. Professional Engineer's license obtained by the successful completion of the PE examination;

TABLE 6.2-1[3]

MINIMUM SHIFT-CREW COMPOSITION[2]

LICENSE CATEGORY	APPLICABLE MODES 1, 2, 3 & 4	5 & 6
Senior Reactor Operator	2	1[1]
Reactor Operator	2	1
Non-Licensed Operator	2	1
Shift Technical Advisor	1[4]	None Required

(1) Does not include the licensed Senior Reactor or Senior Reactor Operator Limited to Fuel Handling individual supervision CORE ALTERATIONS after the initial fuel loading.

(2) The above shift crew composition and the qualified health physics technician of Section 6.2.2 may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided expeditious actions are taken to fill the required position.

(3) Requirements for minimum number of licensed operators on shift during operation in modes other than cold shutdown or refueling are contained in 10CFR50.54(m).

(4) The Shift Technical Advisor position can be filled by either of the two Senior Reactor Operators (a dual-role individual), if he meets the requirements of Specification 6.3.1.b.1.

6.4 TRAINING

A retraining and replacement training program for the facility staff shall be maintained under the direction of the Senior Vice President — Millstone Station and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.59. The Director-Nuclear Training has the overall responsibility for the implementation of the Training Program.

6.5 REVIEW AND AUDIT

6.5.1 Plant Operations Review Committee (PORC)

Function

6.5.1.1 The PORC shall function to advise the Nuclear Unit Director on all matters related to nuclear safety.

Composition

6.5.1.2 The PORC shall be composed of the:

Chairperson:	Nuclear Unit Director
Vice Chairperson & Member:	Operations Manager
Member:	Maintenance Manager
Member:	Instrument and Controls Manager
Member:	Reactor Engineer
Member:	Radiation Protection Supervisor or Chemistry Supervisor
Member:	Engineering Manager
Member:	Staff Engineer

Alternates

6.5.1.3 Alternate members shall be appointed in writing by the PORC Chairperson to serve on a temporary basis; however, no more than two alternates shall participate in PORC activities at any one time.

Meeting Frequency

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairperson.

Quorum

6.5.1.5 A quorum of the PORC shall consist of the Chairperson, or Vice Chairperson, or Senior Vice President — Millstone Station and four members including alternates.

Responsibilities

6.5.1.6 The PORC shall be responsible for:

a. Review of 1) all procedures, except common site procedures, required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Nuclear Unit Director to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Sections 1.0 - 5.0 of these Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and preparation and forwarding of a report covering evaluation and recommendations to prevent recurrence to the Executive Vice President-Nuclear and to the Chairperson of the Nuclear Review Board.

f. Review of all REPORTABLE EVENTS.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews and investigations and reports thereon as requested by the Chairperson of the Nuclear Review Board.

i. Render determinations in writing if any item considered under 6.5.1.6(a) through (d) above, as appropriate and as provided by 10CFR50.59 or 10CFR50.92, constitutes an unreviewed safety question or requires a significant hazards consideration determination.

j. Review of the fire protection program and implementing procedure.

Meeting Frequency

6.5.2.4 The SORC shall meet at least once per six months and as convened by the SORC Chairperson.

Quorum

6.5.2.5 A quorum of the SORC shall consist of the Chairperson or Vice Chairperson and five members including alternates.

Responsibilities

6.5.2.6 The SORC shall be responsible for:

a. Review of 1) all common site procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Senior Vice President — Millstone Station to affect site nuclear safety.

b. Review of all proposed changes to Section 6.0 "Administrative Controls" of these Technical Specifications.

c. Performance of special reviews and investigations and reports as requested by the Chairperson of the Site Nuclear Review Board.

d. Review of the Plant Security Plan and implementing procedures and shall submit changes to the Chairperson of the Site Nuclear Review Board.

e. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairperson of the Site Nuclear Review Board.

f. Review of all common site proposed tests and experiments that affect nuclear safety.

g. Review of all common site proposed changes or modifications to systems or equipment that affect nuclear safety.

h. Render determinations in writing or meeting minutes if any item considered under 6.5.2.6(a) through (g) above, as appropriate and as provided by 10CFR50.59 or 10CFR50.92, constitutes an unreviewed safety question or requires a significant hazards consideration determination.

i. Review of the common site fire protection program and implementing procedures.

Authority

6.5.2.7 The SORC shall:

a. Recommend to the Senior Vice President— Millstone Station written approval or disapproval in meeting minutes of items considered under 6.5.2.6(a) through (g) above.

SPECIAL REPORTS (CONT.)

b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

c. Safety Class 1 Inservice Inspection Program Review, Specification 4.4.10.1.

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Deleted

f. Deleted

g. RCS Overpressure Mitigation, Specification 3.4.9.3.

h. Radiological Effluent Reports required by Specifications 3.11.1.2, 3.11.2.2, 3.11.2.3 and 3.11.4.

i. Degradation of containment structure, Specification 4.6.1.6.4.

j. Steam Generator Tube Inspection, Specification 4.4.5.1.5.

k. Accident Monitoring Instrumentation, Specification 3.3.3.8.

l. Radiation Monitoring Instrumentation, Specification 3.3.3.1.

m. Reactor Coolant System Vents, Specification 3.4.11.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these technical specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to operating procedures.