3/31/76

Docket No. 50-336

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Northeast Nuclear Energy Company
ATTN: Mr. D. C. Switzer
President
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Gentlemen:

The Commission has issued the enclosed Amendment No. 9 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit 2. The amendment consists of changes to the Technical Specifications in response to your applications dated February 10, 1976, February 17, 1976, and March 3, 1976.

The amendment consists of changes in the Technical Specifications that will (1) decrease the set point for reactor trip due to the "Loss of Turbine Hydraulic Fluid Pressure-Low" Reactor Protective System trip from greater than or equal to 1100 psig to greater than or equal to 500 psig, (2) clarify the Surveillance Requirements for the Reactor Protective System "Power Level-High, Nuclear Power", (3) extend the effective date for completion of shoreline protection from March 31, 1976 to June 15, 1976, (4) change an administrative procedure, and (5) allow bypass of the reactor trip function of the Reactor Protective System for "Rate-of-Change of Power-High" above 12% reactor power.

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

Sincerely,

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

Enclosures:
1. Amendment No. 9
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

P

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DATE➤	3/ 18 /76	3/ 18 /76	4/ 5 /76	3/ 31 /76	3/31/76	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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

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 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
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 February 10, 1976, February 17, 1976, and March 3, 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 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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 31, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 9

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace pages I, 1-1, 2-5, 3/4 3-4, 3/4 3-9, 3/4 4-13, 5-6, 6-11, 6-15, 6-17, and 6-18 with the attached revised pages. No changes were made on pages II, 1-2, 2-6, 3/4 3-3, 3/4 3-10, 3/4 4-14, 5-5, 6-12, and 6-16.

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DEFINITIONS

SECTION | PAGE

1.0 DEFINITIONS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2560 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, electric power, cooling or seal water, lubrication or other required auxiliary equipment is also OPERABLE.

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified as a reportable occurrence in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

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

1.8.2 The equipment hatch is closed and sealed, and

1.8.3 The airlock is OPERABLE pursuant to Specification 3.6.1.3.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	$\geq$ 500 psig	$\geq$ 500 psig
11. Rate of Change of Power - High (4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute
12. Steam Generator Water Level - High (5)	$\leq$ 85.40%	$\leq$ 85.40

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 12% of RATED THERMAL POWER.

(5) Each of four channels actuates on the higher of two signals from two downcomer level differential pressure transmitters on each steam generator.

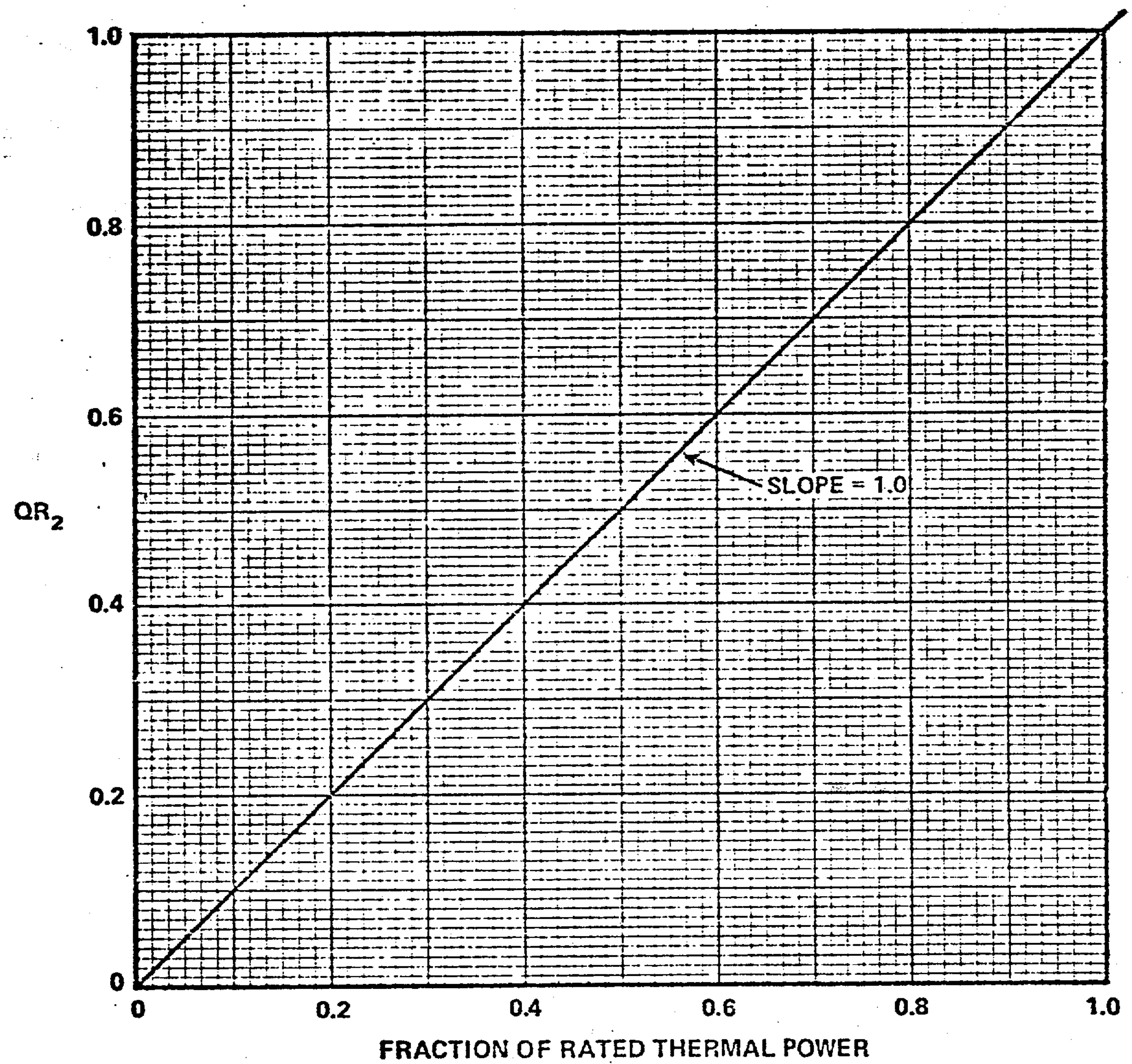

FIGURE 2.2-1

Local Power Density — High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating--Rate of Change of Power - High	4	2(d)	3	1, 2 and *	3
b. Shutdown	4	0	2	3, 4, 5	4
12. Steam Generator Water Level - High	4/SG	2/SG	3/SG	1, 2	3

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(a) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 12% of RATED THERMAL POWER.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) ΔT Power input to trip may be passed below 5% of RATED THERMAL Power; bypass shall be automatically removed when THERMALPOWER is $\geq$ 5% of RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 4 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. $\leq$ 5% of RATED THERMAL POWER, immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

b. > 5% of RATED THERMAL POWER, operation may continue with the inoperable channel in the bypassed condition, provided the following conditions are satisfied:

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Pwr - ΔT Pwr". During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to $\leq$ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The engineered safety feature actuation system instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an engineered safety feature actuation system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either adjust the trip setpoint to be consistent with the value specified in the Trip Setpoint column of Table 3.3-4 within 2 hours or declare the channel inoperable and take the ACTION shown in Table 3.3-3.

b. With an engineered safety feature actuation system instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each engineered safety feature actuation system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ μCi/gram.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with T_{avg} < 515°F within 4 hours.

c. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram, be in HOT STANDBY with T_{avg} < 515°F within 4 hours.

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-2 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

*With $T_{avg} \geq$ 515°F.

ACTION: (Continued)

1. reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. fuel burnup by core region,

3. clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. history of de-gassing operation, if any, and

5. the time duration when the specific activity of the primary coolant exceeded 1.0 μCi/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-2.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,060 + 700/-0 cubic feet.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between assemblies to ensure a $k_{eff} < 0.95$ with the storage pool filled with unborated water.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 22'6".

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I Items in Section 5.1.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 5.8 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower location shall be as shown on Figure 5.1-1.

5.9 SHORELINE PROTECTION

5.9.1 The provisions for shoreline protection described in Amendments 34, 35 and 36 to the FSAR shall be completed by **June 15, 1976.**

REVIEW

6.5.3.7 The NRB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Sections 1.0 - 5.0 of these Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. REPORTABLE OCCURRENCES requiring 24 hour notification to the Commission.

h. Indications of a significant unanticipated deficiency, affecting nuclear safety, in some aspect of design or operation of safety related structures, systems or components.

i. Reports and meetings minutes of the PORC.

AUDITS

6.5.3.8 Audits of facility activities shall be performed under the cognizance of the NRB. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

AUDITS (Continued)

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. Any other area of facility operation considered appropriate by the NRB or the Vice President System Operations.

AUTHORITY

6.5.3.9 The NRB shall report to and advise the Vice President System Operations on those areas of responsibility specified in Sections 6.5.3.7 and 6.5.3.8.

RECORDS

6.5.3.10 Records of NRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NRB meeting shall be prepared, approved and forwarded to the Vice President System Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.3.7 above, shall be prepared, approved and forwarded to the Vice President System Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.3.8 above, shall be forwarded to the Vice President System Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

AUTHORITY

6.5.4.9 The SNRB report to and advise the Vice President System Operations on those areas of responsibility specified in Sections 6.5.4.7 and 6.5.4.8.

RECORDS

6.5.4.10 Records of SNRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SNRB meeting shall be prepared, approved and forwarded to the Vice President System Operations within 14 days following each meeting.

b. Reports of reviews encompassed by, Section 6.5.4.7 above, shall be prepared, approved and forwarded to the Vice President System Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.4.8 above, shall be forwarded to the Vice President System Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PORC and submitted to the NRB and the Superintendent of Nuclear Production.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

SAFETY LIMIT VIOLATION (Continued)

b. The Safety Limit violation shall be reported to the Commission, the Superintendent of Nuclear Production and to the NRB immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NRB and the Superintendent of Nuclear Production within 10 days of the violations.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

b. Refueling operations.

c. Surveillance activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in each document.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PORC and approved by the Plant Superintendent within 7 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations, shall be in accordance with the Regulatory Position in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

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

d. Core Barrel Movement, Specifications 3.4.11 and 4.4.11.

e. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

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 OCCURRENCES submitted to the Commission.

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.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-336

Introduction

On February 10, February 17, and March 3, 1976, Northeast Nuclear Energy Company (NNECO) made applications for license amendment, each of which requested changes to the Technical Specifications for Millstone Unit 2. These NNECO requests are considered herein as a combined licensing action for license amendment.

In the course of reviewing the licensees' applications, it was found that certain changes in the applications would be required. These changes were made after discussion with and the concurrence of the licensee.

Discussion

Descriptions of the requested Technical Specification changes are contained in the following sections.

1. Changes to Technical Specification 2.2.1 - Reactor Trip Setpoints

   a. Loss of Turbine-Hydraulic Pressure-Low

   The turbine generator utilizes an electrohydraulic control (EHC) system which governs the speed, load and flow for startup, operation and shutdown of the turbine. The EHC unit trips the turbine after occurrence of a reactor trip (above 15 percent power), on loss of electrical load, turbine overspeed, low bearing oil pressure, low condenser vacuum, or thrust bearing failure. Low hydraulic oil pressure signals, which are usually indicative of a turbine trip, are supplied from the turbine EHC system to the Reactor Protective System (RPS) to trip the reactor. Technical Specification 2.2-1 gives the RPS "Loss of Turbine-Hydraulic Pressure-Low" as greater than or equal to 1100 psig. NNECO has requested that the set point be lowered to a value greater than or equal to 500 psig which is consistent with the turbine vendor's recommendation; the 1100 psig value was issued in error.

   b. Rate of Change of Power-High

   The RPS is also supplied with a set point which will trip the reactor on "Rate-of-Change of Power-High". Technical Specification 2.2.1 allows bypass of the trip below 10^{-4}% and above 15% power. NNECO has requested that the upper bound of the trip bypass be lowered to a set point at 12% power.

2. Change to Technical Specification 4.3.1.1.1-Reactor Protective Instrumentation

   Specification 4.3.1.1.1 requires that a heat balance be performed on the "Power Level - High (Nuclear)" instrumentation twice daily when the reactor is in Modes 1 or 2, which corresponds to power operation above 5% power and operation at or below 5% power during startup, respectively. The licensee has requested that a heat balance be required only in Mode 1.

3. Change to Technical Specification 5.9.1 - Shoreline Protection

   Specification 5.9.1 requires that provisions for shoreline protection at Millstone Point be completed by March 31, 1976. These provisions consist of shoreline structures which protect against potential damage due to wave scouring. The licensee has requested that the required completion date be extended from March 31, 1976 to June 15, 1976.

4. Change to Technical Specification 6.9.1 - Reporting Requirements

   Specification 6.9.1 requires reports to be submitted in accordance with Title 10, Code of Federal Regulations and Regulatory Guide 1.16, Revision 3. The change updates this requirement to Regulatory Guide 1.16, Revision 4. One of the major features of this change is the substitution of the term "Reportable Occurrence" for "Abnormal Occurrence" in the Technical Specifications. This change and others of similar nature are administrative, do not affect plant safety, and will not be treated further in the following evaluation.

Evaluation

Our evaluation of the licensee's proposed changes is contained in the following sections.

1. Changes to Technical Specification 2.2.1 - Reactor Trip Setpoints

   a. Loss of Turbine Hydraulic Pressure-Low

      With regard to the RPS setpoint for low EHC fluid pressure, in the event of a turbine trip, EHC pressure would fall sufficiently fast so as to trip a setpoint at greater than or equal to 500 psi shortly after one that was set at greater than 1100 psi. Thus, setting this trip at 500 psi will allow the instrumentation to perform its function, i.e., scram the reactor upon turbine trip above 15% power, while preventing reactor scrams due to normal EHC fluid transients. Moreover, the Final Safety Analysis Report (FSAR) for Millstone Unit 2 did not take credit for this setpoint in the "Loss of Load" incident; it serves only as an equipment protection function and is not required to serve a reactor safety function. Accordingly, we find this change acceptable.

b. Rate of Change of Power-High

With regard to the change in setpoint for "High Rate-of-Change of Power" also known as Start-up Rate (SUR), a single bistable switch controls the insertion and/or bypass of three RPS trip input signals. These inputs are derived from: Local Power Density, which must be inserted above 15% power; Turbine Trip/Reactor Trip, which must be inserted above 15% power; and SUR which may be bypassed above 15% power. Of these functions only the Local Power Density was assumed to operate in the accidents considered in Section 14 of the FSAR (Safety Analysis). In order that the Local Power Density trip may be inserted into the RPS circuitry at 15% power, it must be set at approximately 14.5% power in order to account for the bistable dead band. This causes the SUR trip to be inserted at about 12.9% power (decreasing) due to the bistable dead band. Since credit was not taken for the SUR setpoint in the accidents considered in Section 14 of the FSAR, we find this change to be acceptable.

2. Change to Technical Specification 4.3.1.1.1 - Reactor Protective Instrumentation

The licensee has requested a change to the calibration requirements for the "Power Level - High (Nuclear)" instrumentation which provides signal input to the RPS. The change is needed because it is not possible at or below 5% power to obtain adequate heat balance data for accomplishing instrument calibration. We concur with the licensee.

The amended Technical Specification requires that the calibration be carried out above 15% power which provides sufficient heat flux to carry out the required surveillance. We believe that this is the lowest power level at which an accurate heat balance can be performed and is therefore acceptable.

3. Change to Technical Specification 5.9.1 - Shoreline Protection

The licensee has actively pursued completion of the shoreline protective structure described in our Safety Evaluation Report (Supplement No. 2) dated August 1, 1975. Due to unforseen schedule delays, the structure cannot be completed on time and extension of the expected completion date is needed. Historical data indicates that the probability of adverse weather conditions of sufficient magnitude as to cause damage to the shoreline during this extension from March 31, 1976 to June 15, 1976 is acceptably low. Accordingly, we find that the proposed revised completion date of the shore line protection is 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 statement, negative declaration, or 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 change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does 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 31, 1976

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. 9 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 2, located in the Town of Waterford, Connecticut. The amendment is effective as of its date of issuance.

The amendment consists of changes in the Technical Specifications that will (1) decrease the setpoint for reactor trip due to "Loss of Turbine Hydraulic Fluid Pressure-Low" from greater than or equal to 1100 psig to greater than or equal to 500 psig, (2) clarify the Surveillance Requirements for the Reactor Protective System "Power Level-High, Nuclear Power", (3) extend the effective date for the completion of shoreline protection from March 31, 1976 to June 15, 1976, (4) change an administrative procedure, and (5) allow bypass of the reactor trip function of the Reactor Protective System for "Rate of Change of Power-High" above 12% reactor power.

The applications for the amendment complies 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 is 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 statement, negative declaration or 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 February 10, 1976, February 17, 1976, and March 3, 1976, (2) Amendment No. 9 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, Route 156, 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 31 day of March 1976

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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