March 11, 1999

Mr. Martin L. Bowling, Jr.
Recovery Officer - Technical Services
Northeast Nuclear Energy Company
c/o Ms. Patricia A. Loftus
Director - Regulatory Affairs
P. O. Box 128
Waterford, Connecticut 06385

SUBJECT: ISSUANCE OF AMENDMENT - MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 (TAC NO. MA3553)

Dear Mr. Bowling:

The Commission has issued the enclosed Amendment No. 230 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2, in response to your application dated September 9, 1998, as supplemented February 19 and 26, 1999.

The amendment resolves several previously identified technical specifications compliance issues.

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

Sincerely,

ORIGINAL SIGNED BY:
Stephen Dembek, Project Manager
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 230 to DPR-65
2. Safety Evaluation

cc w/encls: See next page

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concur with comments noted

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

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Unit 2

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Millstone Nuclear Power Station
Unit 2

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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. 230
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 September 9, 1998, as supplemented February 19 and 26, 1999, 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, 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 Appendix A, as revised through Amendment No. 230 , 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 11, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 230

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
1-5	1-5
1-6	1-6
3/4 0-1	3/4 0-1
3/4 0-2	3/4 0-2
3/4 2-9	3/4 2-9
3/4 3-25	3/4 3-25
3/4 4-1	3/4 4-1
3/4 4-23	3/4 4-23
B 3/4 0-5	B 3/4 0-5
- -	B 3/4 0-5a
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 3-1a	B 3/4 3-1a
B 3/4 4-7c	B 3/4 4-7c
B 3/4 4-8	B 3/4 4-8

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) used for normal control and indication is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = AY_E + B$$

CORE OPERATING LIMITS REPORT

1.24 The CORE OPERATING LIMITS REPORT is the unit specific document that provides the core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.8. Plant operation within these operating limits is addressed in individual specifications.

ENCLOSURE BUILDING INTEGRITY - DELETED

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.27 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of

ENGINEERED SAFETY FEATURE RESPONSE TIME (Continued)

performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.28 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r

1.29 The TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core. This value includes the effect of AZIMUTHAL POWER TILT.

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

1.31 A RADIOLOGICAL EFFLUENT MONITORING MANUAL shall be a manual containing the site and environmental sampling and analysis programs for measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures to individuals from station operation. An OFFSITE DOSE CALCULATION MANUAL shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints. Requirements of the REMODCM are provided in Specification 6.15.

RADIOACTIVE WASTE TREATMENT SYSTEMS

1.33 RADIOACTIVE WASTE TREATMENT SYSTEMS are those liquid, gaseous and solid waste systems which are required to maintain control over radioactive material in order to meet the LCOs set forth in these specifications.

PURGE - PURGING

1.34 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the containment.

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals, except as provided in LCO 3.0.6. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time it is identified that a Limiting Condition for Operation is not met. Exceptions to these requirements are stated in the individual specifications.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within 2 hours, ACTION shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:

LIMITING CONDITION FOR OPERATION (Continued)

1. At least HOT STANDBY within the next 6 hours.
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This specification is not applicable in MODES 5 or 6.

3.0.6 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance time interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a.

POWER DISTRIBUTION LIMITS

TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F^T_r shall be within the 100% power limit specified in the CORE OPERATING LIMITS REPORT. The F^T_r value shall include the effect of AZIMUTHAL POWER TILT.

APPLICABILITY: MODE 1 with THERMAL POWER >20% RTP*.

ACTION:

With F^T_r exceeding the 100% power limit within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_r to within the power dependent limit specified in the CORE OPERATING LIMITS REPORT and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F^T_r shall be determined to be within the 100% power limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
b. At least once per 31 days of accumulated operation in Mode 1, and
c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.020.

4.2.3.3 F^T_r shall be determined by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump Combination.

*See Special Test Exception 3.10.2

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The coincident logic circuits shall be tested automatically or manually at least once per 31 days. The automatic test feature shall be verified OPERABLE at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or other specified conditions for surveillance testing of the following:

a. Pressurizer Pressure Safety Injection Automatic Actuation Logic; and

b. Pressurizer Pressure Containment Isolation Automatic Actuation Logic; and

c. Steam Generator Pressure Main Steam Line Isolation Automatic Actuation Logic; and

d. Pressurizer Pressure Enclosure Building Filtration Automatic Actuation Logic.

Testing of the automatic actuation logic for Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, and Pressurizer Pressure Enclosure Building Filtration shall be performed within 12 hours after exceeding a pressurizer pressure of 1850 psia in MODE 3. Testing of the automatic actuation logic for Steam Generator Pressure Main Steam Line Isolation shall be performed within 12 hours after exceeding a steam generator pressure of 700 psia in MODE 3.

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

* See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor coolant system vent path consisting of at least two valves in series capable of being powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Reactor Vessel head

b. Pressurizer steam space

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

ACTION:

a. With the Pressurizer vent path inoperable, STARTUP and/or POWER OPERATION may continue provided that i) the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path and ii) one power operated relief valve (PORV) and its associated block valve is OPERABLE; otherwise, restore either the inoperable vent path or one PORV and its associated block valve to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the path to OPERABLE status.

b. With the Reactor Vessel Head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided that the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the Reactor Vessel Head vent path to OPERABLE status within 30 days or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the path to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

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

2. Cycling each valve in the vent path through at least once complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor coolant vent system vent paths during COLD SHUTDOWN or REFUELING.

be consistent with the ACTION statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other divisions must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, action is required in accordance with this specification.

In MODES 5 and 6 Specification 3.0.5 is not applicable, and thus the individual ACTION statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

Specification 3.0.6 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of surveillance requirements to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed surveillance requirements. The Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of equipment being returned to service is reopening a containment isolation valve that has been closed to comply with the Required Actions and must be reopened to perform the surveillance requirements.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of a surveillance requirement on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of a surveillance requirement on another channel in the same trip system.

Specification 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirements. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 This specification establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with two OPERABLE excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits specified in the Core Operating Limits Report using the Power Ratio Recorder. The power dependent limits of the Power Ratio Recorder are less than or equal to the limits specified in the Core Operating Limits Report. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 3) the TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.3.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits specified in the Core Operating Limits Report. The setpoints for these alarms include allowances, set in the conservative directions, for 1) a flux peaking augmentation factor, 2) a measurement-calculational uncertainty factor, 3) an engineering uncertainty factor, 4) an allowance for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor specified in the Core Operating Limits Report. Note the Items (1) and (4) above are only applicable to fuel batches "A" through "L". The Incore Detector Monitoring System is not used to monitor linear heat rate below 20% of RATED THERMAL POWER. The accuracy of the neutron flux information from the incore detectors is not reliable at THERMAL POWER < 20% RATED THERMAL POWER.

3/4.2.3 and 3/4.2.4 TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTORS F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_r and T_q are provided to 1) ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits, and, 2) ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_r or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

Data from the incore detectors are used for determining the measured radial peaking factors. Technical Specification 3.2.3 is not applicable below 20% of RATED THERMAL POWER because the accuracy of the neutron flux information from the incore detectors is not reliable at THERMAL POWER < 20% RATED THERMAL POWER.

The surveillance requirements for verifying that F^T_r and T_q are within their limits provide assurance that the actual values of F^T_r and T_q do not exceed the assumed values. Verifying F^T_r after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.6 DNB MARGIN

The limitations provided in this specification ensure that the assumed margins to DNB are maintained. The limiting values of the parameters in this specification are those assumed as the initial conditions in the accident and transient analyses; therefore, operation must be maintained within the specified limits for the accident and transient analyses to remain valid.

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

ladders, testing one ladder matrix at a time will not remove an RPS channel from the overall logic matrix. Therefore, matrix testing will not remove an RPS channel from service or make the RPS channel inoperable. It is not necessary to enter an action statement while performing matrix testing. This also applies when testing the reactor trip circuit breakers since this test will not remove an RPS channel from service or make the RPS channel inoperable.

The provisions of Specification 4.0.4 are not applicable for the CHANNEL FUNCTIONAL TEST of the Engineered Safety Feature Actuation System automatic actuation logic associated with Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, Steam Generator Pressure Main Steam Line Isolation, and Pressurizer Pressure Enclosure Building Filtration for entry into MODE 3 or other specified conditions. After entering MODE 3, pressurizer pressure and steam generator pressure will be increased and the blocks of the ESF actuations on low pressurizer pressure and low steam generator pressure will be automatically removed. After the blocks have been removed, the CHANNEL FUNCTIONAL TEST of the ESF automatic actuation logic can be performed. The CHANNEL FUNCTIONAL TEST of the ESF automatic actuation logic must be performed within 12 hours after establishing the appropriate plant conditions, and prior to entry into MODE 2.

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. The Reactor Protective and Engineered Safety Feature response times are contained in the Millstone Unit No. 2 Technical Requirements Manual. Changes to the Technical Requirements Manual require a 10CFR50.59 review as well as a review by the Plant Operations Review Committee.

The containment airborne radioactivity monitors (gaseous and particulate) are provided to initiate closure of the containment purge valves upon detection of high radioactivity levels in the containment. Closure of these valves prevents excessive amounts of radioactivity from being released to the environs in the event of an accident. The actuation logic for this function is 1 out of 4. Action Statement 3 of Table 3.3-3 addresses inoperable containment purge channels.

An exception to Technical Specification 3.0.4 is specified for Technical Specification 3.4.9.3 to allow a plant cooldown to MODE 5 if one or both PORVs are inoperable. MODE 5 conditions may be necessary to repair the PORV(s).

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a.

3/4.4.11 Reactor Coolant System Vents

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The flow test verifies that each flowpath through the two solenoid valves is OPERABLE. This verification can be performed by using a series of overlapping tests to ensure flow is verified through all parts of the system.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 230

TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated September 9, 1998, as supplemented February 19 and 26, 1999, the Northeast Nuclear Energy Company, et al. (the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 2 Technical Specifications (TS). The requested changes would change the TS by: (1) changing TS definitions 1.24, "Core Operating Limits Report," 1.27, "Engineering Safety Feature Response Time," and 1.31, "Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)"; (2) changing TS 3.0.2, "Limiting Condition for Operation," by adding a new TS 3.0.6 to the Limiting Condition for Operation TS section; (3) changing TS 4.0.5, "Surveillance Requirements"; (4) changing the mode applicability of TS 3.2.3, "Total Unrodded Integrated Radial Peaking—F_r^T"; (5) changing TS 3.3.2.1, "Engineered Safety Features Actuation System Instrumentation," by modifying TS Table 4.3-2 Table Notation (1) which it references; (6) changing TS 4.4.1.1, "Reactor Coolant System - Coolant Loops and Coolant Circulation Startup and Power Operation;" and, (7) changing TS 4.4.11, "Reactor Coolant System - Reactor Coolant System Vents." The associated TS Bases sections would also be changed. The proposed changes would resolve previously identified TS compliance issues. The supplemental letters provided clarifying information that did not change the original proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice.

2.0 EVALUATION

The licensee has requested changes to the TS to resolve previously identified TS compliance issues. The issues are described below along with the staff's evaluation of the licensee's proposals.

2.1 TS Definitions

Currently, TS definition 1.24, "Core Operating Limits Report," refers to TS 6.9.1.7, "Monthly Operating Report." This reference is in error. The correct reference should be TS 6.9.1.8, "Core Operating Limits Report." The licensee's proposal corrects an editorial error in the TS definition and is therefore acceptable.

Currently, TS definition 1.27, reads, "Engineering Safety Feature Response Time." The word "Engineering" should actually read "Engineered." The licensee's proposal corrects an editorial error in the TS definition and is therefore acceptable.

Currently, TS definition 1.31, "Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)," contains the incorrectly spelled word "radionuclines." The correct spelling should be "radionuclides." Also, the definition's reference to TS 6.16 is incorrect. The correct reference should be to TS 6.15. The licensee's proposal corrects editorial errors in the TS definition and is therefore acceptable.

2.2 Technical Specifications 3.0.2 and 3.0.6

The licensee is proposing to add a new TS, i.e., TS 3.0.6, to allow inoperable equipment to be placed in a condition different from that required by the TS action statement. This new TS will state that it is acceptable to return inoperable equipment to service, under administrative control, but only to demonstrate operability of that equipment, or the operability of other equipment. Since this is an exception to TS 3.0.2, a reference to TS 3.0.6 will be added to TS 3.0.2. The TS Bases will also be changed to reflect the changes to the TS.

The licensee's proposal is consistent with NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," Revision 1, April 1995. The Bases section of NUREG-1432 contains the safety evaluation justifying the proposed TS 3.0.6 wording (the corresponding section in NUREG-1432 is 3.0.5). The applicable portions of NUREG-1432, are quoted below:

> LCO 3.0.[6] establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs [surveillance requirements] to demonstrate:
>
> a. The OPERABILITY of the equipment being returned to service; or
>
> b. The OPERABILITY of other equipment.
>
> The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.
>
> An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

The licensee's proposed TS revision allows the licensee to perform SRs to demonstrate equipment operability. The licensee's proposed TS has previously been shown by the staff, in NUREG-1432, to ensure adequate safety. Therefore, the licensee's proposal is acceptable.

2.3 Technical Specification 4.0.5 and Bases 3/4.4.10

Currently, TS 4.0.5 states, in part:

> Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

Also, TS Bases 3/4.4.10 currently states:

> The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

The licensee is proposing to delete the phrase "(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i)" from TS 4.0.5 and TS Bases 3/4.4.10.

The licensee referenced the guidance of NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," April 1995, for its justification. As noted in NUREG-1482, the NRC staff recognized that situations would arise which would put the licensee in a condition that is not in strict compliance with the TS 4.0.5 requirement to comply with ASME Section XI "except where specific written relief has been granted." For instance, 10 CFR 50.55a(f)(5)(iv) and 10 CFR 50.55a(g)(5)(iv), allows licensees up to 1 year after the start of a new 120-month inspection interval to inform the NRC of code requirements that are impractical. Thus, the licensee's current TS would require them to receive prior NRC approval even though the regulations explicitly allow licensees up to 1 year to inform the NRC of impractical code requirements. The licensee's request is acceptable because continued compliance with the regulation will ensure adequate safety.

2.4 Technical Specification 3.2.3

The licensee is proposing to change the mode of operability for TS 3.2.3, "Total Unrodded Integrated Radial Peaking Factor - F_r^T," from "Mode 1" to "Mode 1 with Thermal Power >20% RTPP*." The licensee stated that the accuracy of the neutron flux information from the incore detectors is not reliable below 20% power. The licensee has proposed that the TS apply only when the data from the incore detectors is reliable. The licensee stated that the current TS surveillance requirements do not require the verification of this limit until prior to operation above 70% following each fuel loading, prior to 31 days accumulated operation in Mode 1, or if the azimuthal power tilt limit is exceeded (TS 3.2.4 which is applicable in Mode 1 above 50% power). Based on the information provided by the licensee, the staff agrees that requiring operability of this instrument during a period when the incoming data is unreliable is not required to ensure safety. Therefore, the staff finds the licensee's proposal acceptable.

The licensee has also proposed several changes to the TS Bases to correct errors and provide more information to the plant operators. These changes are acceptable because they provide the plant operators with more accurate information.

2.5 Technical Specification 3.3.2.1

The licensee is proposing to revise TS 3.3.2.1 to add an exception to TS 4.0.4. The exception will allow a delay in the channel functional test of the automatic actuation logic associated with ESF actuations for safety injection, containment isolation, main steam line isolation, and enclosure building filtration, until the actuation blocks are removed. Normally, the automatic actuation logic for these functions is tested by use of the Automatic Testing Insertor (ATI) circuit. However, the licensee stated that the ATI will not function properly when the features checked by the ATI are blocked or bypassed. During plant startup, the low pressurizer pressure safety injection and the low steam line pressure main steam line isolation actuations are blocked until pressurizer pressure and steam generator pressure have been raised sufficiently to automatically remove the blocks. The pressurizer and steam generator pressures are normally not high enough to remove the blocks until after Mode 3 is entered. The proposed exception to TS 4.0.4 allows entry into Mode 3 with equipment that is inoperable because conditions can not be established to perform the surveillance requirement until after Mode 3 is entered. The applicable pressures at which these actuation blocks are expected to be removed are listed in the new TS.

Based on the information provided by the licensee, the staff finds that the licensee's proposal is acceptable because it ensures the component testing is performed only when the appropriate inputs to the ATI are available, which is consistent with FSAR assumptions for removing engineered safety features actuation system interlocks.

2.6 Technical Specification 4.4.1.1

TS surveillance requirement 4.4.1.1 currently states "The Flow Dependent Selector Switch shall be determined to be in the 4 pump position within 15 minutes prior to making the reactor critical and at least once per 12 hours thereafter." The licensee is proposing to replace this wording with "The above required coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours."

The licensee proposed this change to make the TS SR more consistent with the corresponding Limiting Condition for Operation. The actual requirements for operable reactor coolant pumps and the actual position of the Flow Dependent Selector Switch will not be changed by the licensee's proposal. The licensee's proposal is consistent with NUREG-1432. The Bases section of NUREG-1432 contains the safety evaluation justifying the proposed TS surveillance requirement 4.4.1.1 wording. The applicable portions of NUREG-1432, are quoted below:

> This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

The licensee's proposed TS revision allows the licensee to perform a different SR to demonstrate reactor coolant loop operability. The licensee's proposed TS has previously been shown by the staff, in NUREG-1432, to ensure adequate safety. Therefore, the licensee's proposal is acceptable.

2.7 Technical Specification 4.4.11.3

The licensee is proposing to revise TS SR 4.4.11.3 by deleting the words "during venting," from the sentence "Verifying flow through the reactor coolant vent system vent paths during venting during COLD SHUTDOWN or REFUELING." The licensee requested this change because the current wording requires that flow through the entire reactor vessel head and pressurizer vent paths be verified in Modes 5 and 6. The vent paths discharge through a sparger directly into the containment structure. This will result in possible contamination of the area where the sparger discharges. Additionally, verifying the vent path requires establishment of solid water condition in the reactor coolant system. This could lead to a cold overpressure event.

As an alternative, the licensee is proposing to verify vent flow with a series of overlapping tests. When the overlapping tests are completed, flow will be verified through all parts of the vent system. The licensee's proposal will provide an acceptable alternative method for verifying the vent path while minimizing the potential to contaminate the area surrounding the sparger and minimizing the chance of a reactor coolant system cold overpressure event. Therefore, the licensee's proposal is acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 56251). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S. Dembek

Date: March 11, 1999