April 8, 1999

Mr. Raymond P. Necci
Vice President - Nuclear Oversight and Regulatory Affairs
Northeast Nuclear Energy Company
c/o Mr. David A. Smith
Manager - Regulatory Affairs
P.O. Box 128
Waterford, CT 06385

SUBJECT: ISSUANCE OF AMENDMENT - MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2, RE: LIMITING SAFETY SYSTEM SETTINGS - REACTOR TRIP SETPOINTS (TAC NO. MA4431)

Dear Mr. Necci:

The Commission has issued the enclosed Amendment No. 232 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit 2, in response to your application dated December 28, 1998, as supplemented March 1 and 29, 1999.

The amendment revises Technical Specification 2.2.1, "Limiting Safety System Settings-Reactor Trip Setpoints," to reflect revised loss of normal feedwater flow analyses and authorizes changes to the Final Safety Analysis Report.

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:

Ronald B. Eaton, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

DF01

Docket No. 50-336

Enclosures: 1. Amendment No. 232 to DPR-65
2. Safety Evaluation

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NAME	REaton		TClark		R Bachmann	JClifford				
DATE	04/ 5 /99		04/ 1 /99		04/ 5 /99	04/ 8 /99		02/ /99		02/ /99

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

April 8, 1999

Mr. Raymond P. Necci
Vice President - Nuclear Oversight and Regulatory Affairs
Northeast Nuclear Energy Company
c/o Mr. David A. Smith
Manager - Regulatory Affairs
P.O. Box 128
Waterford, CT 06385

SUBJECT: ISSUANCE OF AMENDMENT - MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2, RE: LIMITING SAFETY SYSTEM SETTINGS - REACTOR TRIP SETPOINTS (TAC NO. MA4431)

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Ronald B. Eaton, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

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2. Safety Evaluation

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Millstone Nuclear Power Station
Unit 2

cc:

Ms. L. M. Cuoco
Senior Nuclear Counsel
Northeast Utilities Service Company
P. O. Box 270
Hartford, CT 06141-0270

Edward L. Wilds, Jr., Ph.D.
Director, Division of Radiation
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

First Selectmen
Town of Waterford
15 Rope Ferry Road
Waterford, CT 06385

Mr. Wayne D. Lanning, Director
Millstone Inspections
Office of the Regional Administrator
475 Allendale Road
King of Prussia, PA 19406-1415

Charles Brinkman, Manager
Washington Nuclear Operations
ABB Combustion Engineering
12300 Twinbrook Pkwy, Suite 330
Rockville, MD 20852

Senior Resident Inspector
Millstone Nuclear Power Station
c/o U.S. Nuclear Regulatory Commission
P.O. Box 513
Niantic, CT 06357

Mr. F. C. Rothen
Vice President - Nuclear Work Services
Northeast Utilities Service Company
P. O. Box 128
Waterford, CT 06385

Ernest C. Hadley, Esquire
1040 B Main Street
P.O. Box 549
West Wareham, MA 02576

Mr. R. P. Necci
Vice President - Nuclear Oversight
and Regulatory Affairs
Northeast Nuclear Energy Company
P. O. Box 128
Waterford, CT 06385

Mr. J. T. Carlin
Vice President - Human Services
Northeast Utilities Service Company
P. O. Box 128
Waterford, CT 06385

Mr. Allan Johanson, Assistant Director
Office of Policy and Management
Policy Development and Planning
Division
450 Capitol Avenue - MS# 52ERN
P. O. Box 341441
Hartford, CT 06134-1441

Mr. M. H. Brothers
Vice President - Millstone Operations
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385

Mr. J. A. Price
Director - Unit 2 Operations
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385

Mr. L. Olivier
Senior Vice President and Chief
Nuclear Officer - Millstone
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385

Millstone Nuclear Power Station
Unit 2

cc:

Citizens Regulatory Commission
ATTN: Ms. Susan Perry Luxton
180 Great Neck Road
Waterford, CT 06385

Deborah Katz, President
Citizens Awareness Network
P. O. Box 83
Shelburne Falls, MA 03170

Ms. Terry Concannon
Co-Chair
Nuclear Energy Advisory Council
Room 4100
Legislative Office Building
Capitol Avenue
Hartford, CT 06106

Mr. Evan W. Woollacott
Co-Chair
Nuclear Energy Advisory Council
128 Terry's Plain Road
Simsbury, CT 06070

Little Harbor Consultants, Inc.
Millstone - ITPOP Project Office
P. O. Box 0630
Niantic, CT 06357-0630

Attorney Nicholas J. Scobbo, Jr.
Ferriter, Scobbo, Caruso, Rodophele, PC
1 Beacon Street, 11th Floor
Boston, MA 02108

Mr. D. B. Amerine
Vice President - Engineering Services
Northeast Nuclear Energy Company
P. O. Box 128
Waterford, CT 06385

Mr. D. A. Smith
Manager - Regulatory Affairs
Northeast Nuclear Energy Company
P. O. Box 128
Waterford, CT 06385

Ms. Nancy Burton
147 Cross Highway
Redding Ridge, CT 00870

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. 232
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 28, 1998, as supplemented March 1 and 29, 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. 232 , 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

James W. Clifford

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 8, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 232

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
2-4	2-4
B 2-6	B 2-6
B 2-7	B 2-7

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Level-High		
	Four Reactor Coolant Pumps Operating	$\leq$ 9.6% above THERMAL POWER, with a minimum setpoint of $\leq$ 14.6% of RATED THERMAL POWER, and a maximum of $\leq$ 106.6% of RATED THERMAL POWER.	$\leq$ 9.7% Above THERMAL POWER, with a minimum of $\leq$ 14.7% of RATED THERMAL POWER, and a maximum of $\leq$ 106.7% of RATED THERMAL POWER.
3.	Reactor Coolant Flow - Low (1)	$\geq$ 91.7% of reactor coolant flow with 4 pumps operating*.	$\geq$ 90.9% of reactor coolant flow with 4 pumps operating.
4.	Reactor Coolant Pump Speed - Low (1)	$\geq$ 830 rpm	$\geq$ 823 rpm
5.	Pressurizer Pressure - High	$\leq$ 2397 psia	$\leq$ 2407 psia
6.	Containment Pressure - High	$\leq$ 4.42 psig	$\leq$ 5.07 psig
7.	Steam Generator Pressure - Low (2) (5)	$\geq$ 691 psia	$\geq$ 677 psia
8.	Steam Generator Water Level - Low (5)	$\geq$ 48.5% Water Level - each steam generator	$\geq$ 47.5% Water Level - each steam generator
9.	Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

*Design Reactor Coolant flow with 4 pumps operating is the lesser of either:
a. The reactor coolant flow rate measured per Specification 4.2.6.1, or
b. The minimum value specified in the CORE OPERATING LIMITS REPORT.

Steam Generator Water Level - Low

The Steam Generator Water Level-Low Trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded.

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.17.

Thermal Margin/Low Pressure (Continued)

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1865 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2.25°F to compensate for potential temperature measurement uncertainty; and a further allowance of 74 psi to compensate for pressure measurement error, trip system processing error, and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 74 psi allowance is made up of a 5 psi bias, a 19 psi pressure measurement allowance and a 50 psi time delay allowance.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuring transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 232

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 December 28, 1998, as supplemented March 1 and 29, 1999, the Northeast Nuclear Energy Company, et al. (NNECO, or the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit 2, Technical Specification (TS) regarding revised loss of normal feedwater analyses. Specifically, the reactor trip setpoints for low steam generator level, in TS Table 2.2-1, would be revised. The setpoint change results in an earlier reactor trip on decreasing steam generator level. In addition to the TS changes, the licensee has also modified the loss of normal feedwater (LONF) transient analysis in both chapter 10 and chapter 14 of the plant Final Safety Analysis Report (FSAR). These modifications were needed to permit lower auxiliary feedwater flow which is now, in part, offset by the earlier reactor trip signal. The supplemental submittals provided additional information that did not change the staff's proposed no significant hazards consideration determination.

2.0 EVALUATION

TS Table 2.2-1

A change to TS Table 2.2-1 has been requested to raise the reactor trip setpoint for steam generator water level to 48.5%, with an allowable value of 47.5%. The licensee has justified these values by performing an analysis with the trip assumed to occur at 43%. The analysis assumes the trip setpoint is lower than the TS setpoint to account for the appropriate setpoint and instrumentation uncertainties. The change results in the reactor trip on reducing steam generator level initiating earlier than it does now. The reason for the change is to partially offset a reduction in auxiliary feedwater (AFW) flow. The licensee has performed the transient analysis with the modified setpoints with the AFW flow and concluded that the acceptance criteria continues to be met.

Although increasing the setpoint may increase the likelihood of a reactor trip on steam generator level, the licensee does not expect the setpoint to be approached during normal plant operation and has stated that an unexpected plant event would be needed to cause the setpoint to be reached. Additionally, the licensee has adjusted the pretrip alarm in the control room to provide the operators with the same advanced notice of a steam generator low level condition. The staff finds the proposed changes acceptable.

FSAR Chapter 10

The licensee has modified FSAR Chapter 10. The modifications include a reference to a new best estimate of LONF analysis. The licensee has stated that the revised analysis now credits the atmospheric dump valves (ADVs) in lieu of the main steam safety valves to remove heat from the generator. Crediting the ADVs results in increased flow to the steam generators because the ADVs can be opened at lower pressure and the AFW system delivers more water to the steam generators at reduced pressure. The staff has determined that crediting the ADVs for the FSAR Chapter 10 analysis is acceptable. With the credit for the ADVs the licensee has stated that the loss of feedwater design basis continues to be met. As a result, the staff finds the proposed changes to be acceptable.

FSAR Chapter 14

The licensee has performed a reanalysis of the FSAR Chapter 14 LONF transient analysis. The analysis was performed at the new setpoints and reduced AFW, and shows acceptable results. In addition to the changes to the flow and setpoints, the licensee has made a number of other changes to the transient analysis. The analysis shows that for the most limiting LONF cases analyzed, assuming a single failure, the steam generators do not empty, the pressurizer does not go water solid, and the steam generators do not exceed 110% of the design pressure. The licensee has stated that another decrease in heat removal from the secondary system event, the loss of electric load or turbine trip event, continues to be more limiting from both the standpoint of minimum departure from nucleate boiling ratio (DNBR) and from a peak reactor coolant system (RCS) standpoint. As a result, these aspects of the LONF event do not need to be evaluated.

In the performance of the new analysis, the licensee has used a different NRC-approved evaluation model. The methodology is contained in the report ANF-89-151(P)(A) ANF-RELAP METHODOLOGY FOR PRESSURIZED WATER REACTORS: ANALYSIS OF NON-LOCA CHAPTER 15 EVENTS, and was approved by the staff in March of 1992. The methodology is appropriate for evaluating the LONF event. The licensee has analyzed five different cases to determine the most limiting conditions. The cases analyzed were chosen to maximize pressurizer water level and minimum steam generator water level and considered different combinations of the limiting single failures and the availability of offsite power. The limiting single failure was either a motor driven or turbine driven AFW pump. The analysis now credits automatic initiation of the turbine driven pump and conservatively assumes a minimum total AFW flow that includes instrument uncertainties and a 5 percent pump degradation. The initial conditions were also biased to maximize pressurizer water level and minimize steam generator water level. The initial steam generator and reactivity feedback values were conservatively selected in accordance with the approved topical report. The licensee also considered both the availability and unavailability of the normal plant controls and offsite power to be assured the limiting event was considered.

The licensee has also changed the way the main steam safety relief valve accumulation is modeled in the new analysis. By letter dated March 29, 1999, the licensee stated that rather than assuming the valve opens at the nominal setpoint plus 3 percent to account for drift, plus another 3 percent to account for accumulation, the licensee has modeled the valves to open at the nominal setpoint plus 3 percent to account for drift with a 0.1-second delay to account for valve accumulation. The licensee has justified this assumption by referencing a statement by the valve vendor which stated that the valve will go full open in about 20 to 30 milliseconds. Because the 0.1 second time assumed in the analysis is much higher than 20 to 30 milliseconds, the staff finds the modified assumption to be acceptable.

The analysis results for all cases show that the pressurizer does not go water solid, the steam generators do not empty, and decay heat is removed without exceeding 110 percent of the main design pressure. Because the licensee has demonstrated acceptable results for the LONF event by performing a reanalysis with an NRC-approved evaluation model that considers the limiting single failure and conservatively modeled the plant and the initial conditions, the staff finds the changes to be acceptable.

Instrument Setpoint Methodology

In a letter dated December 28, 1998, the licensee stated that the instrument setpoint methodology used to calculate trip setpoints and allowable values is consistent with the approach used in TS changes previously submitted by letters dated July 21 and October 6, 1998. The staff has previously reviewed these TS changes and has determined that the licensee's setpoint methodology is consistent with the guidance of Regulatory Guide 1.105, Rev. 2 and ISA Standard 67.04, 1982 and, is therefore, acceptable. The staff finds that the licensee's methodology for the proposed changes is acceptable. However, the licensee did not account for the effects of harsh environment on instrument drift because this instrumentation is not subjected to harsh environment. Feedwater system pipebreaks inside and outside containment are not included in the licensing basis for this plant (FSAR Section 14.2.8). In a conference call with the licensee, the staff requested the licensee to confirm that reactor trip on low SG water level has not been credited in any other event resulting in a harsh environment. In a letter dated March 1, 1999, the licensee documented that this instrumentation will not be required to trip the reactor when subjected to a harsh environment. Based on this documentation, the staff finds the FSAR and TS changes related to reactor trip on low SG water level to be acceptable.

The licensee has also revised the FSAR and TS Bases Section 2.2.1 for thermal margin/low pressure reactor trip setpoint from 1850 psia to 1865 psia to account for the uncertainties caused by the harsh environment. The staff finds the proposed change to be acceptable because it properly accounts for the uncertainties caused by the harsh environment.

Credit for Auto Initiation of Auxiliary Feedwater Start Signal

The revised analysis of LONF, documented in the licensee's submittal dated December 28, 1998, takes credit for the automatic initiation of Motor Driven Auxiliary Feedwater Pumps, within 4 minutes, after steam generator water level reaches the automatic auxiliary feedwater actuation setpoint. FSAR Section 7.3, "Engineered Safety Features Actuation System", which includes the Auxiliary Feedwater Automatic Initiation System (AFAIS), states that this system is designed to meet the requirements of Institute of Electrical and Electronics Engineers (IEEE)

Standard 279-1971, "Criteria for Nuclear Generating Station Protection Systems." Therefore, since this system meets the requirements of IEEE-279, the licensee can take credit for the system in the LONF event.

The licensee included conforming Bases pages with the amendment request. The NRC does not review and approve the Bases but they are included to maintain a current Authority File.

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 requirements with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20. 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 (64 FR 6701, February 10, 1999). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

Principal Contributors: C. Jackson
H. Garg
V. Ordaz

Date: April 8, 1999