June 7, 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: CORRECTION OF SAFETY EVALUATION IN 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 Amendment No. 232 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit 2, in a letter dated April 8, 1999.

The Safety Evaluation (SE) contained a typographical error on page two of the SE. Please substitute the enclosed corrected page to your amendment and any distribution copies you my have made.

I regret any inconvenience this my have caused you. If you have any questions regarding this correction please call me on (301) 415-3041.

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

Docket No. 50-336

Enclosure: As Stated

cc w/encl: See next page

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

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

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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

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 to 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 motor 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.