April 14, 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 (TAC NO. MA4578)

Dear Mr. Necci:

The Commission has issued the enclosed Amendment No. 234 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit 2, in response to your application dated January 18, 1999.

The amendment revises Technical Specification (TS) 3.6.1.2, "Containment Systems - Containment Leakage," and also revises the related TS bases and Final Safety Analysis Report sections. The revisions relate to changes in the secondary containment bypass leakage.

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

Docket No. 50-336

Enclosures: 1. Amendment No. 234 to DPR-65
2. Safety Evaluation

cc w/encls: See next page

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

April 14, 1999

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Northeast Nuclear Energy Company
c/o Mr. David A. Smith
Manager - Regulatory Affairs
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Docket No. 50-336

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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. 234
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 January 18, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 234 , 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, Section Chief
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 14, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 234

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
3/4 6-2	3/4 6-2
3/4 6-5	3/4 6-5
B 3/4 6-1	B 3/4 6-1
-- --	B 3/4 6-1a

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of < L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 54 psig.

b. A combined leakage rate of < 0.60 L_a for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

c. A combined leakage rate of < 0.0072 L_a for all penetrations that are secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.0072 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in accordance with the Containment Leakage Rate Testing Program.

This Page Intentionally Deleted.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

Primary CONTAINMENT INTEGRITY is required in MODES 1 through 4. This requires an OPERABLE containment automatic isolation valve system. In MODES 1, 2, and 3 this is satisfied by the automatic containment isolation signals generated by low pressurizer pressure and high containment pressure. In MODE 4 the automatic containment isolation signals generated by low pressurizer pressure and high containment pressure are not required to be OPERABLE. Automatic actuation of the containment isolation system in MODE 4 is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating engineered safety features components. Since the manual actuation (trip pushbuttons) portion of the containment isolation system is required to be OPERABLE in MODE 4, the plant operators can use the manual pushbuttons to rapidly position all automatic containment isolation valves to the required accident position. Therefore, the containment isolation trip pushbuttons satisfy the requirement for an OPERABLE containment automatic isolation valve system in MODE 4.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to < 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is in accordance with the Containment Leakage Rate Testing Program.

The Millstone Unit No. 2 FSAR contains a list of the containment penetrations that have been identified as secondary containment bypass leakage paths.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and

3.6.1.2. The limitations on the air locks allow entry and exit into and out of the containment during operation and ensure through the surveillance testing that air lock leakage will not become excessive through continuous usage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 234

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 January 18, 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 Specifications (TS) and Final Safety Analysis Report (FSAR) regarding the secondary containment bypass leakage.

2.0 BACKGROUND

This application proposed the following changes to TS 3.6.1.2, *Containment Systems - Containment Leakage:*

The maximum allowable secondary containment bypass leakage provided in TS 3.6.1.2.c will be reduced from $<0.017 \times L_a$ to $<0.0072 \times L_a$. This change is being implemented for consistency with the leak rate assumed in the design basis loss-of-coolant accident (LOCA).

TS Table 3.6.1.2 Table 3.6-1, *Secondary Containment Bypass Leakage Paths*, will be removed.
TS 3.6.1.2.c will be revised to remove the reference to Table 3.6-1.

The bases for TS 3.6.1.2 will be revised to add a reference to the FSAR for a list of the containment penetrations that have been identified as secondary containment bypass leakage paths.

Additionally, the FSAR will be revised, as discussed below, to reflect the changes to the TS:

FSAR Section 5.3.4 will be revised to include the additional secondary containment bypass leakage paths that have been identified and the criteria under which they were identified.

The current discussion in Section 5.3.4 regarding the different bypass leakage assumption used in the control room dose calculations will be deleted. The control room doses have been recalculated using the bypass value specified in the revised TS 3.6.1.2.

The LOCA analysis description and results in Section 14.8.4 have been updated for consistency with the revised analyses.

The licensee's request includes the removal of information from the TS. Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including:

(1) safety limits, limiting safety system settings and limiting control settings;

(2) limiting conditions for operation;

(3) surveillance requirements;

(4) design features; and

(5) administrative controls.

On July 19, 1995, the Commission published revisions to 10 CFR 50.36 specifying what must be included in limiting conditions for operation (LCO) in the TS (60 FR 36953). The four criteria added to 10 CFR 50.36 for determining whether a particular matter is required to be included in the TS, are as follows:

(1) Installed instrumentation that is used to detect, and indicate in the control room, a significant
abnormal degradation of the reactor coolant pressure boundary;

(2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier;

(3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; and

(4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

As a result, existing TS LCOs which fall within or satisfy any of the criteria in 10 CFR 50.36 must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents.

3.0 EVALUATION

3.1 Proposed Revision to Maximum Allowable Secondary Containment Bypass Leakage

The original analysis of record for offsite doses from a design basis accident (DBA) LOCA assumed a secondary containment bypass leakage rate of $0.017 \times L_a$. The original analysis of record for control room doses from a DBA LOCA assumed a secondary containment bypass leakage rate of 11 cc/hr. NNECO updated the DBA LOCA evaluation to reflect changes in several analysis assumptions and inputs. A description of the assumptions, inputs, methods used, and results obtained were submitted in a licensee amendment request dated September 28, 1998. As a result of an NRC request for information dated December 3, 1998, NNECO agreed to revise the LOCA analysis to use the same secondary containment bypass leakage rate for both the offsite and control room dose assessments. A description of the analysis assumptions, inputs, methods, and results obtained was submitted in a letter dated January 20, 1999. In performing this re-analysis, NNECO determined that it would be necessary to reduce the secondary containment bypass leakage from $<0.017 \times L_a$ to $<0.0072 \times L_a$.

The current license amendment request would revise TS 3.6.1.2.c. to reflect the reduced bypass leakage value assumed in the revised LOCA analysis. The NRC reviewed the revised LOCA analysis as part of its review of the September 28, 1998, license amendment request, as supplemented by letters dated January 7 and January 20, 1999. The results obtained by NNECO showed reduced offsite doses, but increased control room doses. The NRC staff granted the licensee's request in a March 10, 1999, license amendment (Amendment No. 228). In the safety evaluation for that licensing action, the NRC staff concluded that, in the event of a LOCA at Millstone Nuclear Power Station Unit 2, the offsite radiation doses would be within the dose guidelines of 10 CFR Part 100 and would be acceptable. The NRC staff also concluded that the Millstone Nuclear Power Station Unit 2, control room doses would be within the acceptance criteria of 10 CFR Part 50, Appendix A, GDC-19, and NUREG-0800 Section 6.4 and would be acceptable. On this basis, the staff concludes that the proposed revision to TS 3.6.1.2.c. to reduce the specified bypass leakage from $<0.017 \times L_a$ to $<0.0072 \times L_a$ is acceptable.

3.2 Removal of Table 3.6-1; Updating Bases

Relocating the list of secondary containment bypass paths from TS 3.6.1.2 to the FSAR and the supporting change to the bases of the TS can have no impact on the postulated consequences or probability of analyzed accidents. The relocated requirements do not meet any of the four criteria in 10 CFR 50.36 and can be removed and relocated to a licensee-controlled document, in this case the FSAR. It is not necessary to maintain a list of the secondary containment bypass leakage paths in the TS. Therefore, the staff concludes that this change is acceptable.

3.3 Inclusion of Additional Bypass Paths and Identification Criteria in the FSAR

The NRC staff has reviewed the proposed revision to FSAR page 5.3-7. Since the maximum allowable secondary containment bypass leakage is controlled by TS, the FSAR discussion of additional bypass paths will not increase the consequence of previously analyzed design basis accidents. The NRC staff has reviewed the identification criteria proposed by NNECO. NNECO proposed that for a leakage pathway to viably result in bypass leakage, the pathway must be open to the containment atmosphere post-accident and must provide a means of transporting the containment atmosphere beyond the enclosure building filtration region as well as a means for the containment atmosphere to escape the piping or duct. The staff finds that this criterion is appropriate given that the intent of limiting secondary containment bypass leakage is to minimize the unfiltered release of primary containment airborne activity to the environment.

3.4 FSAR Changes

The NRC staff reviewed the proposed FSAR pages for the LOCA in its review of the license amendment request dated September 28, 1998, with supplements dated January 7 and January 20, 1999, and found those revisions to be acceptable. The NRC staff granted the licensee's request in a March 10, 1999, license amendment (Amendment No. 228).

3.5 Evaluation Summary

Based on its review of the information provided by NNECO related to the proposed FSAR and TS changes, the NRC staff finds reasonable assurance that the radiological consequences of anticipated accidents at Millstone Nuclear Power Station, Unit 2 will continue to be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A, GDC-19 and Section 6.4 of NUREG-0800. Therefore, the proposed changes are acceptable.

4.0 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.

5.0 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 6703, 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.

6.0 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: S. LaVie
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Date: April 14, 1999