July 2, 1999

Mr. R. P. Necci - Vice President
Nuclear Oversight and Regulatory Affairs
c/o Mr. David A. Smith
Northeast Nuclear Energy Company
P. O. Box 128
Waterford, CT 06385-0128

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE RUPTURE ACCIDENT ANALYSIS (TAC NO. MA2021)

Dear Mr. Necci:

The Commission has issued the enclosed Amendment No. 172 to Facility Operating License No. NPF-49 for the Millstone Nuclear Power Station, Unit No. 3 (MP3), in response to your application dated June 5, 1998, as supplemented by a letter dated January 13, 1999.

The amendment revises the MP3 licensing basis associated with the design basis steam generator tube rupture (SGTR) accident analysis described in Chapter 15 of the MP3 Final Safety Analysis Report (FSAR) to address an unreviewed safety question. The SGTR analyses described in the FSAR include an offsite dose analysis and a margin to overfill analysis. Both of these analyses were updated. The offsite dose analysis was updated to reflect a larger capacity for the steam generator atmospheric dump valve (ADV) and a decreased operator response time to close the ADV block valve, and the margin to overfill analysis was updated to reflect a new single failure. The revised licensing basis will be incorporated into the FSAR and will revise the SGTR accident analysis to address the changes in the offsite dose and margin to overfill analyses.

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:
John A. Nakoski, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 172 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

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

July 2, 1999

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Nuclear Oversight and Regulatory Affairs
c/o Mr. David A. Smith
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Docket No. 50-423

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2. Safety Evaluation

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

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. NPF-49

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 June 5, 1998, as supplemented by a letter dated January 13, 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, changes to the Final Safety Analysis Report (FSAR) to reflect the description of the revised steam generator tube rupture accident analysis to address the changes in the offsite dose and margin to overfill analyses as set forth in the application for amendment by the licensee, dated June 5, 1998, and supplemented by a letter dated January 13, 1999, are authorized.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

B.C. Buckley for

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

Date of Issuance: July 2, 1999

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated June 5, 1998, as supplemented by a letter dated January 13, 1999, the Northeast Nuclear Energy Company, et al. (the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 3 (MP3), licensing basis. The requested changes would revise the MP3 licensing basis associated with the design basis steam generator tube rupture (SGTR) accident analysis described in Chapter 15 of the MP3 Final Safety Analysis Report (FSAR) to address an unreviewed safety question. The SGTR analyses described in the FSAR include an offsite dose analysis and a margin to overfill analysis. Both of these analyses were updated. The offsite dose analysis was updated to reflect a larger capacity for the steam generator atmospheric dump valve (ADV) and a decreased operator response time to close the ADV block valve, and the margin to overfill analysis was updated to reflect a new single failure on the atmospheric dump bypass valves (ADBV) that could cause failure to open on demand of the ADBV associated with two of the intact steam generators. The revised licensing basis will be incorporated into the FSAR and will revise the SGTR accident analysis to address the changes in the offsite dose and margin to overfill analyses. The January 13, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee's revised SGTR analysis uses the same methodology as the current analysis. This methodology is described in WCAP-10698 "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," March 30, 1987, that was approved by the staff. The revised analysis for radiological consequences assumes a larger ADV flow capacity of 820,000 lb/hr/valve. However, the operator action time for the closure of the block valve to isolate the stuck-open ADV on the ruptured steam generator is assumed to be 20 minutes in the revised analysis instead of 30 minutes as assumed in the current analysis. The quicker closure of the block valve more than compensates for the larger capacity assumed for the ADV. The results of the revised analysis show that the radiological consequences are not increased from those previously calculated.

The revised analysis of a margin to steam generator (SG) overfill assumes a most limiting single failure that causes failure to open of the ADBV associated with two of the intact steam generators (SGs) when the operator initiates cooling of the reactor coolant system using the intact SGs. In this part of the analysis, a larger minimum capacity for the ADBV of 820,000 lb/hr/valve is credited. The licensee states that this larger flow capacity is still a conservative minimum capacity for the ADBV. The operator actions and their limiting time credited in the SG overfill analysis are not being changed. The results of the licensee's revised analysis shows that sufficient margin still exists for SG overfill following a design basis SGTR event.

Operator Response Times

The staff's evaluation of WCAP-10698 stipulates plant-specific criteria for assessing operator response times in the event of an SGTR. The staff used those criteria to evaluate the information submitted by the licensee regarding operator response times during an SGTR at MP3. The staff based its evaluation on the June 5, 1998, submittal, as supplemented in the January 13, 1999, letter.

Criterion 1: Provide simulator and emergency operating procedure (EOP) training related to a potential SGTR.

The licensee documented by letter dated January 13, 1999, that onsite simulator and EOP training relevant to an SGTR are provided. The staff finds that the licensee has satisfied Criterion 1.

Criterion 2: Utilizing typical control room staff, complete demonstration runs to show that the postulated SGTR accident can be mitigated within a period of time compatible with overfill prevention.

By letter dated January 13, 1999, the licensee submitted the demonstrated operator response times for the overfill scenario. The demonstrated response times, compared with the times assumed in the licensee's SGTR analysis, are given in the following table:

Operator Action	Demonstrated Time* (minutes:seconds)	Assumed Time (minutes)
Event initiation (malfunction inserted) to **feed flow to ruptured SG stopped** (E-3, step 4.b)	17:12	16.5
Feed flow to ruptured SG stopped (E-3, step 4.b) to **reactor coolant system (RCS) cooldown initiated** (E-3, step 14.b)	6:21	8.0
RCS cooldown terminated (E-3, step 14.d) to **RCS depressurization initiated** (E-3, step 18.b)	1:07	3.0
RCS depressurization terminated (E-3, step 18.c) to all but one charging pump stopped (E-3, step 21)	1:29	3.0
Total	26:09	30.5

*The demonstrated response time was derived from the arithmetic mean of response times from all crews, both operations and administration crews.

Further, the licensee indicated that simulation runs were completed with three administration crews and five operations crews. The staff derived the arithmetic mean for operator response times, which were used as the demonstrated times (shown in the preceding table) for the overfill scenario. The licensee stated in its January 13, 1999, submittal that one operations crew failed to meet the total assumed time of 30.5 minutes. The licensee also stated that during retesting this crew was successful on two different occasions, with total demonstrated times of 24 and 20 minutes. The average demonstrated operator response time (17 minutes and 12 seconds) for "feed flow to ruptured SG stopped" was 42 seconds greater than the time assumed (16 minutes and 30 seconds) in the licensee's SGTR analysis. The staff found this discrepancy to be acceptable because the average total operator response time (26 minutes and 9 seconds was bounded by the total assumed time (30 minutes and 30 seconds).

In addition, the licensee's January 13, 1999, submittal presented information on the control room staff's ability to (1) determine that the ruptured SG's ADV had failed open, and (2) complete actions to close the block valve of the ruptured steam generator's ADV within the required time of 20 minutes. The licensee stated that each ADV has indicating lights that show whether the valve is open or closed. The licensee also stated that there is an annunciator on Main Board 5, "Main Steam Relief Valve Not Closed," that annunciates when an ADV opens. To close the block valve of the ruptured SG's ADV, the operator must push the "close" button and check the position indication to verify that the valve closed. Finally, the licensee stated that during informal sessions two crews closed the subject block valves in less than 10 minutes. Crews will be formally timed in the 1999/2000 licensed operator requalification training cycle. The staff finds this information to be acceptable.

On the basis of the information discussed, the staff finds that the licensee has satisfied Criterion 2. The staff also finds that the licensee has given satisfactory assurance that operators can identify that the ruptured SG's ADV is open, and then can close the associated block valve within the required time of 20 minutes.

Criterion 3: If the emergency operating procedures (EOPs) specify SG sampling as a means of identifying the SG with the ruptured tube, provide the expected time period for obtaining the sample results and discuss the effect on the duration of the accident.

By letter dated January 13, 1999, the licensee stated that SG sampling is performed daily. The licensee noted that counting the sample and identifying which generator has a ruptured tube would take about 1 hour. The licensee explained that the leak rate can be estimated by sampling the condensate polishing facility effluent and steam jet air ejector. The licensee indicated that this sampling can be performed concurrent with the SG sampling and is also estimated to take about 1 hour. The licensee stated that SG tube leakage so small that the sampling that would be required to identify the SG with the ruptured tube would be bounded by the analysis performed for the complete severance of one SG tube for both the overfill and dose effects. In addition, the licensee stated that if an unexpected increase in SG level has not occurred and sampling is required to identify the SG with the ruptured tube, then steam generator overfill is not a concern. On the basis of this information, the staff finds that Criterion 3 is satisfied.

Dose Assessment

The staff reviewed the inputs and assumptions of the licensee's offsite dose calculation and found the licensee's analysis to be acceptable. The staff performed a confirmatory analysis using the licensee's assumptions for the SGTR and calculated comparable results. Staff analysis assumptions and results are presented below, along with the licensee's results. Staff dose calculations also assumed the 20-minute closure time for the affected SG ADV block valves proposed by the licensee and found to be acceptable by the staff as previously discussed. The licensee's calculated offsite dose consequence results meet the acceptance criteria in 10 CFR 100, and thus are acceptable. The current FSAR control room dose is evaluated for the limiting loss-of-coolant accident (LOCA). The requested changes to the SGTR dose analysis offset one another such that the calculated offsite doses decrease. For this case, the staff concludes that the postulated dose to operators in the control room would likely also decrease. Therefore, the staff did not evaluate the control room dose due to the revised SGTR analysis. The staff finds the proposed change to be acceptable with regard to the radiological consequences of the SGTR.

Offsite Radiation Dose Results:	Dose to Thyroid (rem)		
	Licensee Calculated	Staff Calculated	10 CFR 100 Acceptance Criteria
1. Accident Initiated Iodine Spike			
Exclusion Area Boundary	19	18	30
Low Population Zone	2.0	1.4	30
2. Pre-accident Iodine Spike			
Exclusion Area Boundary	51	50	300
Low Population Zone	4.0	3.6	300

Staff Assumptions for SGTR Analysis:

Iodine species	100% elemental
Reactor Coolant Iodine Activity	
Accident Initiated Spike	1.0 µCi/gm of Dose Equivalent (D.E.) I-131, with iodine spike 500 times larger at assumed appearance rates given (see Tables 1 and 2 below)
Preaccident Spike	60 µCi/gm of D.E. I-131 (see Table 1)
Secondary System Initial Activity	0.1 mCi/gm of D.E. I-131 (see Table 1)

Table 1 - Iodine Activities in Primary and Secondary Coolant

	Activity (Ci)		
	Primary Coolant D.E. I-131		Secondary Coolant D.E. I-131
Nuclide	1.0 μCi/gm	60 μCi/gm	0.1 μCi/gm (per SG)
I-131	108	10804	3.36
I-132	63.5	3798	1.12
I-133	283	16985	4.79
I-134	39.6	2383	0.204
I-135	152	9153	2.10

Table 2 - Iodine Spike Appearance Rates

Nuclide	Ci/sec	Ci/hr
I-131	1.70	6120
I-132	3.24	11664
I-133	3.84	13680
I-134	4.71	16956
I-135	3.59	12924

Mass water in RCS (lbm)		520,000
Mass water initially in each SG (lbm)		95,170
Break Flow (0-2 hrs) (lbm)		211,400
Loss of offsite power at time of reactor trip (sec)		109
Faulted SG ADV Block valve closed (min)		20
Primary-to-secondary leakage (hrs)		8
Total primary-to-secondary leakage (gpm)		1.0
Atmospheric Dispersion Factors		
EAB X/Q (s/m^3)	0 - 2 hrs	4.3 E-04
LPZ X/Q (s/m^3)	0 - 8 hrs	2.9 E-05
Activity release data from submittal		Table 15.6.3-3, Figure 15.6.3-6, and Figure 15.6.3-12
Iodine partition coefficient		0.01

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. 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 35992). 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: Michelle Hart
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Date: July 2, 1999