January 31, 1995

Mr. John F. Opeka
Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, CT 06141-0270

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M90038)

Dear Mr. Opeka:

The Commission has issued the enclosed Amendment No. 102 to Facility Operating License No. NPF-49 for the Millstone Nuclear Power Station, Unit No. 3, in response to your application dated July 22, 1994.

The amendment revises the Technical Specifications to incorporate a different setpoint and transient methodology for determining the maximum allowable power range neutron flux setpoint. These changes allow Millstone Unit 3 to operate with a reduced number of main steam-line safety valves at a reduced power level, as determined by the high flux setpoint.

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

Sincerely,

Original signed by:

Vernon L. Rooney, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 102 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

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Docket File	MGriggs	OC/LFDCB
PUBLIC	VRooney	LNicholson, RGI
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DOCUMENT NAME: G:\GRIGGS\M90038.AMD

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

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Docket No. 50-423

Enclosures: 1. Amendment No. 102to NPF-49
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cc w/encls: See next page

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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. 102
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 July 22, 1994, 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. NPF-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 102 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip F. McKee Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 31, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3/4 7-2	3/4 7-2
B 3/4 7-1	B 3/4 7-1
B 3/4 7-2	B 3/4 7-2
---	B 3/4 7-2a

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	65
2	46
3	28

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR*	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	47
2	33
3	19

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The design minimum total relieving capacity for all valves on all of the steam lines is 1.579×10^7 lbs/h which is 105% of the total secondary steam flow of 1.504×10^7 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$Hi\ \phi = (100/Q)\ \frac{(w_s h_{fg} N)}{K}$$

where:

$Hi\phi$ = Safety Analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{\text{(Btu/sec)}}{\text{Mwt}}$

h_{fg} = heat of vaporization for steam at the highest MSSV opening pressure including tolerance (± 3%) and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

SAFETY VALVES (Continued)

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs. The following plant specific safety valve flow rates were used:

SG Safety Valve Number (Bank No.)	Main Steam System	
	Set Pressure (psia)	Flow (lbm/hr per loop)
1	1200	893,160
2	1210	900,607
3	1220	908,055
4	1230	915,502
5	1240	922,950

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating or accident conditions coincident with a total loss-of-offsite power.

The auxiliary feedwater system is capable of delivering a total feedwater flow of 480 gpm at a pressure of 1236 psia to the entrance of at least three steam generators while allowing for (1) any spillage through the design worst-case break of the Normal feedwater line, (2) the design worst-case single failure; and (3) recirculation flow. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed into operation.

SAFETY VALVES (Continued)

3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK

The OPERABILITY of the demineralized water storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 10 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power, and with an additional 6-hour cooldown period to reduce reactor coolant temperature to 350°F. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 102

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 July 22, 1994, the Northeast Nuclear Energy Company (the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 3 Technical Specifications (TS). The requested changes revise the Technical Specifications to incorporate a different setpoint and transient methodology for determining the maximum allowable power range neutron flux setpoint. These changes allow Millstone Unit 3 to operate with a reduced number of main steam-line safety valves at a reduced power level, as determined by the high flux setpoint.

2.0 BACKGROUND

On August 22, 1994, the U.S. Nuclear Regulatory Commission (NRC) issued an information notice (IN 94-60) to alert all holders of operating licenses or construction permits for pressurized water reactors. The notice cautioned the addressees about a potential for overpressurizing the main steam system during periods when one or more main steam safety valves (MSSV) are inoperable. All recipients were expected to review the information for applicability to their facilities and to consider actions to avoid such problems.

Westinghouse Nuclear Safety Advisory Letter NSAL-94-001, "Operation at Reduced Power Levels With Inoperable MSSVs," January 20, 1994, describes a deficiency in the basis for Standard Technical Specifications (STS) Table 3.7.1, "Operable Main Steam Safety Valves Versus Applicable Power in Percent of Rated Power." The basis assumes that the maximum allowable initial power level is a linear function of main steam safety valve flow capacity. Westinghouse has determined that this assumption is not correct and notes that, when operating at low power in accordance with STS Table 3.7.1, with one or more safety valves inoperable, a loss-of-load/turbine trip (LOL/TT) transient concurrent with the loss of feedwater could result in overpressurization of the main steam system.

The equation used in the original Westinghouse bases for Table 3.7.1 reduces the power range neutron flux trip setpoint linearly with the assumed reduction in safety valve flow capacity. Therefore the potential for overpressurization of the main steam system exists. NSAL 94-001 recommends the use of a more conservative equation to calculate the power range high neutron flux trip setpoint. The new equation is linear with respect to the power level rather than with respect to safety valve flow.

3.0 EVALUATION

The licensee proposed changes that would revise the TS Table 3.7-1 and associated Bases by changing the maximum allowable power range neutron flux high setpoint when one or more MSSVs are inoperable. New values for TS Table 3.7.1 were submitted and the corrected methodology was provided to the NRC staff for review. The licensee calculated the new neutron flux high setpoints using the Westinghouse revised algorithm.

The principal reason for this change is that in calculating allowed power levels, transient response based on less than full power operating conditions was not accounted for properly. When the Millstone Unit 3 plant is operating at reduced power levels, a high pressurizer pressure or overtemperature ΔT trip will be delayed or may not occur following a LOL/TT with concurrent loss of main feedwater which is the design basis transient for determining the MSSV capacity. Under those conditions, the reactor trip may be initiated by a signal from low steam generator water level at a later time into the transient which may cause the secondary system pressure to exceed 110% of the design pressure.

The NRC staff has evaluated the proposed method and agrees that the allowable power level calculated by this conservative method would bound the design basis transient initiated at all power levels and would not lead to overpressurization in the secondary system. The new setpoints are more conservative than the previous setpoints. Based on the above information, the NRC staff approves the proposed changes to TS Table 3.7-1 and has no objection to changes made in Bases 3/4.7.1.1.

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 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 (59 FR 47171). 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 Contributor: Maudette Griggs

Date: January 31, 1995