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Docket No. 50-336

SEP 5 1984

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Mr. W. G. Counsil, Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 97 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit 2, in response to your application dated October 12, 1983 as supplemented May 16, 1984.

This amendment changes the Technical Specifications to:

- Revise the pressurizer level band to a wider range during periods of normal operation, and

- Impose more restrictive operability requirements for the pressurizer heaters.

A copy of our Safety Evaluation is enclosed. The notice of issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

Original signed by:
D. B. Osborne, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 97 to DPR-65
2. Safety Evaluation

ORB#3:DL	ORB#3:DL	ORB#3:DL	OELD	AD:OR:DL
PKreutzer	DBOsborne:dd	JRMiller	Bachmann	GCLainas
8/22/84	8/23/84	8/ /84	8/29/84	8/31/84

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Northeast Nuclear Energy Company

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

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. 97
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 October 12, 1983 as supplemented May 16, 1984 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 (1) 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 Appendices A and B, as revised through Amendment No. 97 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 5, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 97

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Remove and 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 area of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 4-4	3/4 4-4
B 3/4 4-2	B 3/4 4-2

REACTOR COOLANT SYSTEM

RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3 Two power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 8 hours either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 8 hours either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.1 Each PORV shall be demonstrated OPERABLE:

a. Once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. Once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.3.2 Each block valve shall be demonstrated OPERABLE once per 92 days by operating the valve through one complete cycle of full travel. This demonstration is not required if a PORV block valve is closed and power removed to meet Specification 3.4.3 a or b.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4. The pressurizer shall be OPERABLE with:

a. A water volume greater than or equal to 525 cubic feet (35%) but less than or equal to 1050 cubic feet (70%), and

b. At least two groups of pressurizer heaters each having a capacity of at least 130 kW.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients.

A single reactor coolant loop with its steam generator filled above 10% of the span provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cool-down if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

In MODES 4 and 5, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs $\leq 275^{\circ}F$ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than $43^{\circ}F$ ($31^{\circ}F$ when measured by a surface contact instrument) above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 296,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

3/4.4.3 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.4 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the reactor coolant system during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which occurs if the heaters are energized uncovered. The maximum water level in the pressurizer ensures that this paramter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish and maintain natural circulation.

The requirement that 130 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of off-site power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL;

MILLSTONE NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-336

Introduction

By letter dated October 12, 1983, (Ref. 1) Northeast Nuclear Energy Company (NNECO or licensee) proposed two changes to the Millstone Nuclear Power Station, Unit 2, Technical Specification 3.4.4, entitled Pressurizer. The first change revises the allowed pressurizer level band during operation in Modes 1, 2 and 3. The current Technical Specification 3.4.4 requires the pressurizer level to be maintained within $\pm$ 5% of its programmed value during periods of normal operation. The proposed modification allows the pressurizer level to be maintained between 35% and 70%, inclusive. The licensee proposed this change in order to allow for more effective pressurizer cooling for entry into Mode 4. Additionally, since the pressurizer level change is used in the method for determining reactor coolant system leakage, the proposed revision will allow more data to be obtained on the rate of pressurizer level decrease.

The second change to the Technical Specification (TS) imposes more restrictive operability requirements for the pressurizer heaters. The current TS requires the operability of at least 130kw of pressurizer heater capacity powered from emergency power supplies. Should the heaters become inoperable, the licensee has 72 hours to restore the emergency power supply or be in at least Hot Standby within 6 hours and Hot Shutdown within 12 hours. The proposed change requires the operability of at least two groups of pressurizer heaters, each with a capacity of a least 130kw, which are capable of being supplied by emergency power. If one of these groups becomes inoperable, the current Action Statement, described above, is employed. If both groups become inoperable, the unit must be placed in Hot Standby within 6 hours and Hot Shutdown within 12 hours.

Evaluation

To assure that the proposed modification to the pressurizer level band does not significantly affect the consequences of postulated transients and accidents, the licensee reviewed the plant safety analyses and assessed the impact of the proposed change on the event consequences. The licensee's evaluations are documented in references 1 and 2.

The licensee assessed the impact of the proposed pressurizer level change on overheating transients by reanalyzing the limiting transients, the loss of load and loss of normal feedwater transients, with a 75% pressurizer level.

These analyses were performed with the LOFTRAN code and were compared to the results documented in the Basic Safety Report (BSR) for Millstone Unit 2 Cycle 4 operation. For the loss of load event, the reanalysis showed a peak pressure of 2581 psia as compared to the BSR value of 2573 psia. For the loss of normal feedwater event, peak pressure was 2538 psia using the 75% pressurizer level. Minimum DNBR was greater than 1.30 for both events. As neither case violated the acceptance criterion for peak pressure of 2750 psia (110% of design pressure), nor did they violate the minimum DNBR criterion, the staff finds the event consequences acceptable.

The effect of the proposed change on overcooling events was assessed by examining the steam line rupture accident which is the limiting overcooling event. The BSR analysis was performed using 31% pressurizer level and the results showed that the minimum DNBR was greater than 1.3. As this event was calculated using a pressurizer level which is less than that proposed by the revised TS and results in acceptable consequences, the staff finds that the proposed change in pressurizer level will not significantly affect plant consequences for overcooling events.

Evaluation of the effect of the proposed pressurizer level change was also performed for the SG tube rupture and small break LOCA events. The effect of the proposed change in pressurizer level on the SG tube rupture event is to delay the reactor trip on low pressurizer pressure and thereby increase mass released through the tube rupture. This event was previously analyzed using a pressurizer level of 65% and was reanalyzed by the licensee using a pressurizer level of 70%. The reanalysis showed there was no significant impact on the transient. The staff finds this assessment acceptable.

For the small break LOCA, the worst case break, a 0.1 ft^2 break in the pump discharge piping, was re-evaluated using a pressurizer level of 35%. The results showed that the pressurizer would empty 20 seconds earlier and the consequent minimum core inventory and peak cladding temperature would occur 20 seconds sooner than the previously analyzed case. The earlier core uncovery results in an increase in the cladding temperature of 14°F to 1985°F, thus meeting the peak cladding temperature limit of 2200°F as specified by 10 CFR 50.46. Hand calculations have been performed which verified the licensee's conclusions that the pressurizer would drain approximately 20 seconds earlier. Thus, the staff finds the results acceptable.

For other postulated transients and accidents, the licensee concluded that the proposed change in pressurizer level band would not impact the results. Based on our review of the BSR, the staff concurs with the licensee's assessment.

Relative to the proposed change in the operability requirements to the pressurizer heaters, the staff finds the change to be acceptable as it is more restrictive than currently employed.

Based upon the foregoing, the staff has concluded that the proposed changes to TS 3.4.4, entitled Pressurizer, are acceptable.

Environmental Consideration

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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 this amendment.

Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: September 5, 1984.

Principal Contributor:

R. Jones, RSB

References

1. Letter, W. G. Council (NNECO) to J. R. Miller (NRC), "Millstone Nuclear Power Station, Unit No. 2, Proposed Revisions to Technical Specification Modification of Pressurizer Level Band," October 12, 1983.

2. Letter, W. G. Council (NNECO) to J. R. Miller (NRC) "Millstone Nuclear Power Station, Unit No. 2 Additional Information to Support Modification of the Pressurizer Level Band," May 16, 1984.