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Docket No. 50-336

SEP 1 2 1977

Northeast Nuclear Energy Company
ATTN: Mr. D. C. Switzer, President
P. O. Box 270
Hartford, Connecticut 06101

Gentlemen:

The Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application dated March 25, 1977.

The amendment modifies the existing Technical Specifications 3.4.11 and 4.4.11.2 which address the required use of the Internal Vibration Monitor (IVM).

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 31
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

Const. 1
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subject to modifications noted on "working copy"

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NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

THE CONNECTICUT LIGHT AND POWER COMPANY,
THE HARTFORD ELECTRIC LIGHT COMPANY,
WESTERN MASSACHUSETTS ELECTRIC COMPANY, AND
NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. DPR-65

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company (the licensees), dated March 25, 1977, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 31, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 12, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 31

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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 also provided to maintain document completeness.

Pages

3/4 4-32
3/4 4-33
B 3/4 4-12

TABLE 4.4-4 (Cont'd)

11. Out of core monitor welds prevent 100% accessibility to the vessel circumferential welds for preservice examination. Approximately 80-90% of the circumferential weld area shall be examined for the preservice examination.

12. Laterial restraints for the three support nozzles were redesigned to improve the accessibility to the nozzle-to-shell welds in the area of the support feet. Improved accessibility afforded by this redesign coupled with the design of the support feet, will allow approximately 70-80% of the support nozzle-to-shell welds to be examined.

13. The preoperational (baseline) examinations shall include all of the designated welds for both steam generators. The extent and frequency of inservice examinations shall be as listed in Table ISC-251 and Section ISC-240 of the Winter 1972 Addendum to Section XI of the ASME Code.

14. Each reactor coolant pump flywheel shall be inspected to the maximum extent practical as described below:

    a. An in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway at approximately 3-1/3 year intervals, during the refueling or maintenance shutdowns coinciding with the inservice inspection schedule as required by ASME Boiler and Pressure Vessel Code, Section XI.

    b. A complete ultrasonic examination of all exposed surfaces at approximately 10-year intervals, during the plant shutdown coinciding with the inservice inspection schedule required by the ASME Boiler and Pressure Vessel Code, Section XI. Removal of the flywheels is not required to perform these examinations.

    c. Examination procedure and acceptance criteria to be in conformance with the requirements equivalent to those specified for Class I vessels in the ASME Boiler and Pressure Vessel Code, Section III - Nuclear Power Plant Components. All flywheels shall be inspected by the same volumetric examination methods used during the baseline examination.

CORE BARREL MOVEMENT

LIMITING CONDITION FOR OPERATION

3.4.11 Core barrel movement shall be limited to less than the Amplitude Probability Distribution (APD) and Spectral Analysis (SA) Alert Levels for the applicable THERMAL POWER level.

APPLICABILITY: MODE 1.

ACTION:

a. With the APD exceeding its applicable Alert Level, measure and process a spectral analysis within 48 hours. The APD Alert and Action Levels may be adjusted to reflect changes in RMS noise level resulting from neutronic effects.

b. With the SA exceeding its applicable Alert Levels, POWER OPERATION may proceed provided the following actions are taken:

   1. APD shall be measured and processed at least once per 24 hours.

   2. SA shall be measured within 48 hours and at least once per 7 days thereafter; SA shall be processed at least once per 7 days. The APD Alert and Action Levels may be adjusted to reflect changes in RMS noise level resulting from neutronic effects.

   3. A Special Report, identifying the cause(s) for exceeding the applicable Alert Level, shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days of detection.

c. With the APD and/or SA exceeding their applicable Action Levels, measure and process APD and SA data within 24 hours to determine if the core barrel motion is exceeding its limits. With the core barrel motion exceeding its limits, reduce the core barrel motion to within its Action Levels within the next 24 hours or be in HOT STANDBY within the following 6 hours.

ACTION: (Continued)

d. With the APD and/or SA data processing equipment inoperable, perform the required monitoring and data processing within the next 7 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Baseline Monitoring Core barrel movement Alert Levels and Action Levels, as determined by APD and SA monitoring of the excore neutron detectors, shall be determined at nominal THERMAL POWER levels of 20%, 50%, 80% and 100% of RATED THERMAL POWER during the reactor startup test program; these Alert Levels and Action Levels shall be reported in a Special Report pursuant to Specification 6.9.2 within 31 days after initially reaching 100% of RATED THERMAL POWER.

4.4.11.2 Routine Monitoring Core barrel movement shall be determined to be less than the APD and SA Alert Levels by using the excore neutron detectors to measure APD and SA at the following frequencies:

a. APD data shall be measured and processed at least once per 7 days.

b. SA data shall be measured and processed at least once per 31 days.

for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The required inspection programs for the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent practicable, the inspection program for the Reactor Coolant System components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code "Inservice Inspection of Nuclear Reactor Coolant Systems" dated July 1, 1971.

All areas scheduled for volumetric examination have been pre-service examined using equipment, techniques and procedures anticipated for use during post-operation examinations. To assure that consideration is given to the use of new or improved inspection equipment, techniques and procedures, the Inservice Inspection Program will be periodically reviewed on a 5 year basis.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most reactor coolant loop components except the reactor vessel. The reactor vessel requires special consideration because of the radiation levels.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts.

BASES

The nondestructive testing for repairs on components greater than 4 inches diameter gives a high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. Repairs on components 4 inches in diameter or smaller receive a surface examination which assures a similar standard of integrity. In each case, the leak test will ensure leak tightness during normal operation.

For normal opening and reclosing, the structural integrity of the Reactor Coolant System is unchanged. Therefore, satisfactory performance of a system leak test at 2250 psia following each opening and subsequent reclosing is acceptable demonstration of the system's structural integrity. These leak tests will be conducted within the pressure-temperature limitations for Inservice Leak and Hydrostatic Testing of Specification 3.4.9.1 and Figure 3.4-2.

Inspection of the pipe hangers and supports provides assurance that these devices are operated within permissible travel and/or loading limits.

3/4.4.11 CORE BARREL MOVEMENT

This specification is provided to ensure early detection of excessive core barrel movement if it should occur. Neutron noise levels are used to continually monitor core support barrel (CSB) motion. Change in motion is manifested as changes in the four excore neutron detector signals. Baseline core barrel movement Alert Levels and Action Levels at nominal THERMAL POWER levels of 20%, 50%, 80% and 100% of RATED THERMAL POWER will be determined during the reactor startup test program.

Data from these detectors is to be reduced in two forms. RMS values are computed from the Amplitude Probability Density (APD) of the signal amplitude. These RMS magnitudes include variations due both to various neutronic effects and internals motion. Consequently, these signals alone can only provide a gross measure of CSB motion. A more accurate assessment of CSB motion is obtained from the Auto and Cross Power Spectral Densities (PSD, XPSD), phase (ϕ) and coherence (COH) of these signals. These data result from a Spectral Analysis (SA) of the excore detector signals.

A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of the first fuel cycle.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

Introduction

By application for license amendment dated March 25, 1977, Northeast Nuclear Energy Company (NNECO) requested a change to the Millstone Unit No. 2 Technical Specifications. The proposed change would modify the existing Technical Specifications 3.4.11 and 4.4.11.2 which address the required use of the Internals Vibration Monitor (IVM). In the process of reviewing NNECO's application of March 25, 1977, we have found it necessary to make changes to the proposed Technical Specifications. These changes have been discussed with and concurred in by NNECO

Background

Millstone Unit No. 2 is equipped with an Internals Vibration Monitor (IVM) which is an automated computer-based system designed to acquire, analyze, and interpret the variation in data from the four excore nuclear detector strings. The purpose of the system is to identify reactor vessel internals motion, specifically core barrel movement.

Data acquired from the IVM is processed into two forms (output functions). The first form, the Amplitude Probability Density (APD), represents an approximate quantitative measure of vibrational amplitude. The second form, the Spectral Analysis (SA) indicates the relative importance of the various vibrational modes displayed by the reactor internals. The proposed Technical Specifications require NNECO to take various actions in the event that the APD or SA exceed predetermined levels as measured from their initial or "baseline" values. These actions are referred to as Alert and Action levels.

Alert and Action levels for the APD and SA data were chosen during and as one outcome of a baseline monitoring program conducted prior to initial reactor power operation at Millstone Unit No. 2. The APD and SA data were both based upon the Core Support Barrel - reactor vessel (CSB-RV) snubber gap variations (movements).

The Alert and Action levels for each of the APD and SA are the same. The Alert Levels are set at one-third of the snubber gap for the APD and SA which gives more than 99% confidence that the peak vibrational amplitude does not exceed the CSB-RV snubber gap. The Action Level for the APD and SA is set at two-thirds of the snubber gap which corresponds to a confidence level greater than 85% that the CSB-RV gap is not exceeded by the vibrational amplitude. If the vibrational amplitude were to exceed the CSB-RV gap, the CSB would impact the reactor vessel wall causing possible damage to the reactor internals.

Discussion and Evaluation

A discussion and our evaluation of the proposed changes to the Millstone Unit No. 2 Technical Specifications is contained in the following sections:

1. Changes to Technical Specification 3.4.11 - Limiting Conditions for Operation for Core Barrel Movement

   Barrel Movement

   Three significant changes are proposed for Section 3.4.11. These changes provide for (a) replacement of the predetermined Alert and Action levels for the APD with adjustable levels; (b) deletion of a requirement which would cause a Special Report to be submitted in the event that the APD and/or the SA exceed their baseline level by 10%; and (c) an additional provision to allow for the inoperability of the APD and/or SA data processing equipment. These changes are discussed below.

   a. Adjustable Alert and Action Levels for the APD

      Operating experience has indicated that the output of the IVM system is affected by changes in core conditions exclusive of changes in core barrel movement. Neutronic noises can cause the overall "noise" level to increase without any increase in core barrel movement. Based on industry data to date, the magnitude of the neutronic effects increases with core average fuel burnup (MWD/MTU). This phenomenon appears to be influenced by the change in soluble boron concentration of the primary coolant, i.e., the lower the boron concentration, the larger the neutronic effects.

      After adjustment of the APD Alert and Action Levels, the confidence that the peak vibrational amplitude will not exceed the CSB-RV snubber gap will remain the same. This is because the basis for these levels is still the actual measured snubber gap clearance. The adjustments in the Alert and Action Levels account for the fact that a portion of the neutron noise signal appears to change with boron concentration.

      Based upon the above, we conclude that the proposed revisions to Technical Specifications 3.4.11a and 3.4.11b.2, which allow the APD Alert and Action levels to be adjusted will not affect the probability of occurrence, consequences, or detectability of core barrel motion and are therefore acceptable.

b. Deletion of the Reporting Requirement -

APD and/or SA Exceeding 10% of "Baseline" Values

We have reevaluated the staff position which had been the basis for requiring a Special Report to be submitted in the event that the APD and/or the SA exceeded 10% of the "baseline" values. Based upon operating experience of licensees, we find that a 10% increase in the APD or SA is not unusual and does not represent a significant increase. Accordingly, we find that deletion of this reporting requirement, contained in Technical Specification 3.4.11c, is acceptable.

c. Additional Provision to Allow for Inoperability of the APD and/or SA Data Processing Equipment

Technical Specification 4.4.11.2 requires that APD and SA data be processed on a routine basis; however, the existing Technical Specifications do not provide any time for remedial action to be taken in the event that routine monitoring cannot be completed in the specified allowable time interval due to inoperability of data processing equipment.

NNECO has proposed that if the APD and/or the processing equipment is inoperable, the required monitoring must be completed within 7 days or the reactor must be placed in Hot Standby (neutron multiplication factor less than 0.99 and the average coolant temperature $\geq$ 300°F) within the next 6 hours.

We have reviewed the above proposed Technical Specification and we conclude that 7 days would be sufficient time to allow for alternate means to process APD and/or SA data. Moreover, any significant increase in core barrel movement, as indicated by changes in the APD and SA, would take place over a period of time considerably longer than 7 days. In this regard, in the period from January 1976 to September 1977, the increase in CSB-RV vibrational amplitude at Millstone Unit No. 2 was less than 10% of the CSV-RV snubber gap. Therefore, an equipment outage which prevents or delays processing APD or SA data for 7 days will not significantly affect the ability of NNECO to monitor the level of core barrel movement. Accordingly, the provisions of Technical Specification 3.4.11d are acceptable.

2. Changes to Technical Specification 4.4.11.2 - Surveillance Requirements for Core Barrel Movement

The existing requirements of Technical Specification 4.4.11.2a call for the APD data to be taken every 24 hours. NNECO has proposed that the 24 hour interval for APD data acquisition be extended to 7 days. Based upon our evaluation presented in Section 1. above, we concluded that not analyzing APD and/or SA data for 7 days would not significantly affect the ability of NNECO to monitor core barrel movement. Accordingly, the proposed change to Technical Specification 4.4.11.2a, which extends the APD data acquisition interval from 24 hours to 7 days is acceptable.

In the event that the APD/SA processing equippment became inoperable following the proposed seven day surveillance interval, a cumulative interval of up to 14 days could elapse before the APD/SA processing equipment would be required to be operable or the plant be put in the Hot Standby condition. In view of our previous conclusion regarding the time for increase in CSB-RV vibrational amplitude, this 14 day period will not result in a significant increase in CSB-RV vibrational amplitude and is still accpetable.

Environmental Considerations

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: September 12, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

NORTHEAST NUCLEAR ENERGY COMPANY,
THE CONNECTICUT LIGHT AND POWER COMPANY,
THE HARTFORD ELECTRIC LIGHT COMPANY, AND
WESTERN MASSACHUSETTS ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 31 to Facility Operating License No. DPR-65 issued to Northeast Nuclear Energy Company, The Connecticut Light and Power Company, The Hartford Electric Light Company, and Western Massachusetts Electric Company, which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2, located in the Town of Waterford, Connecticut. The amendment is effective as of the date of issuance.

The amendment modifies the existing Technical Specifications 3.4.11 and 4.4.11.2 which address the required use of the Internals Vibration Monitor (IVM).

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to

10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 25, 1977, (2) Amendment No. 31 to License No. DPR-65, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Waterford Public Library, Rope Ferry Road, Waterford, Connecticut 06385. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 12th day of September 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors