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PDR

The Commission has issued the enclosed Amendment No. 49 to Facility Operating License No. DFR-65 for Millstone Nuclear Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated January 12 and March 4, 1982 as modified by your letters of January 22, February 5, 12 and March 1, 1982.

The changes to the TS allow reactor heatup for Cycle 5 operation with:

- o Less than 100 percent inspection of the tubes in each steam generator (SG) based on the results of the tube inspection program reported in your application; and
- o Additional reporting requirements on primary to secondary leakage rates.

Some portions of your proposed TS have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff. The Cycle 5 reload amendment will be issued under separate cover.

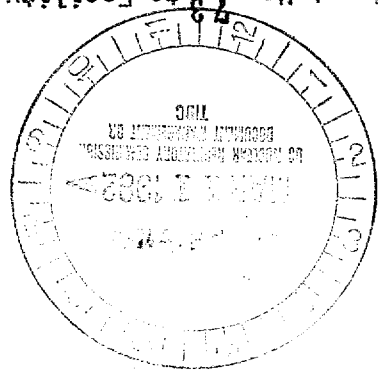
We herein acknowledge your agreement, as stated in your March 4, 1982 letter to:

- o Provide the results of the profilometry inspections at the egg crate supports and your assessment of the results within 60 days following startup from the outage; and
- o Inspect the 22 steam generator tubes in the sludge pile area that were not inspected in 1982 during the next ECT inspection.

1
CP

Mr. W. G. Counsil, Senior Vice President
Nuclear Engineering & Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:



Docket No. 50-336

*about file
DCS MS-016*

MAR 5 1982

A copy of the Safety Evaluation and the related Notice of Issuance are also enclosed.

Sincerely,

Original signed by:

Eben L. Conner, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:

- 1. Amendment No. 73 to DPR-65
- 2. Safety Evaluation
- 3. Notice of Issuance

cc w/enclosures:
See next page

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| | <i>102-11/3/82</i> | <i>102-11/3/82</i> | <i>102-11/3/82</i> | <i>102-11/3/82</i> | <i>102-11/3/82</i> | <i>102-11/3/82</i> |
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| SURNAME | PMKreutzer | <i>ES</i> | RAClark | WVJohnston | MNovak | M. Rothschil |
| DATE | 3/3/82 | 3/3/82 | 3/3/82 | 3/4/82 | 3/3/82 | 3/04/82 |

Concurs in lic. amendment & Fed. Reg. note subject to corrections noted.



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

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Docket File
ORB#3 Rdg
PMKreutzer

Docket No. 50-336

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: NORTHEAST NUCLEAR ENERGY COMPANY, ET AL. Millstone Nuclear
Power Station, Unit No. 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- Notice of Availability of Applicant's Environmental Report.
- Notice of Proposed Issuance of Amendment to Facility Operating License.
- Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- Notice of Availability of NRC Draft/Final Environmental Statement.
- Notice of Limited Work Authorization.
- Notice of Availability of Safety Evaluation Report.
- Notice of Issuance of Construction Permit(s).
- Notice of Issuance of Facility Operating License(s) or Amendment(s).

Other: Amendment No. 73.
Referenced documents has been provided PDR.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

| | | | | | | |
|-----------|---------------|--|--|--|--|--|
| OFFICE → | ORB#3:DR | | | | | |
| SURNAME → | PMKreutzer/ph | | | | | |
| DATE → | 3/5/82 | | | | | |

Northeast Nuclear Energy Company

cc:

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Day, Berry & Howard
Counselors at Law
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Hartford, Connecticut 06103

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Manager - Washington Nuclear
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Waterford Public Library
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Waterford, Connecticut 06385

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Representative
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Boston, Massachusetts 02203

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P. O. Drawer KK
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Regional Administrator
Nuclear Regulatory Commission, Region I
Office of Inspection and Enforcement
631 Park Avenue
King of Prussia, Pennsylvania 19406

cc w/enclosure(s) and incoming
dtd: 1/12/82, 1/ 15/82

Office of Policy & Management
ATTN: Under Secretary Energy
Division
80 Washington Street
Hartford, Connecticut 06115



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

NORTHEAST NUCLEAR ENERGY COMPANY
THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
THE WESTERN MASSACHUSETTS ELECTRIC COMPANY
DOCKET NO. 50-336
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
License No. DPR-65

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendments by Northeast Nuclear Energy Company (the licensee) dated January 12 and March 4, 1982, as supplemented by letters dated January 22, February 5, 12 and March 1, 1982, 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 applications, 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. 73, 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



Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: MAR 5 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 73

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 also provided to maintain document completeness.

Remove

3/4 4-7g
3/4 4-7h

Replace

3/4 4-7f
3/4 7-8

TABLE 4.4-6

STEAM GENERATOR TUBE INSPECTION

| 1ST SAMPLE INSPECTION | | | 2ND SAMPLE INSPECTION | | 3RD SAMPLE INSPECTION | |
|--------------------------------------|--------|---|---|--|-----------------------|---|
| Sample Size | Result | Action Required | Result | Action Required | Result | Action Required |
| A minimum of S Tubes per S. G. | C-1 | None | N/A | N/A | N/A | N/A |
| | C-2 | Plug defective tubes and inspect additional 2S tubes in this S. G. | C-1 | None | N/A | N/A |
| | | | C-2 | Plug defective tubes and inspect additional 4S tubes in this S. G. | C-1 | None |
| | | | | | C-2 | Plug defective tubes |
| | | | | | C-3 | Perform action for C-3 result of first sample |
| | C-3 | Perform action for C-3 result of first sample | N/A | N/A | | |
| | C-3 | Inspect all tubes in this S. G., plug de- fective tubes and inspect 2S tubes in each other S. G. * Prompt notification to NRC pursuant to specification 6.9.1 | All other S. G.s are C-1 | None | N/A | N/A |
| | | | Some S. G.s C-2 but no additional S. G. are C-3 | Perform action for C-2 result of second sample | N/A | N/A |
| | | | Additional S. G. is C-3 | Inspect all tubes in each S. G. and plug defective tubes. * Prompt notification to NRC pursuant to specification 6.9.1 | N/A | N/A |

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

*For the Cycle 5 refueling outage only, action required for result C-3 is to inspect all tubes in the affected area of both steam generators as identified in Figures 1 through 4 of the licensee's submittal of February 12, 1982. The affected area of each tube is between the tube sheet and the first tube support in both hot and cold leg sides.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

- a. A containment atmosphere particulate radioactivity monitoring system,
- b. The containment sump level monitoring system, and
- c. A containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- a. With one of the above radioactivity monitoring leakage detection systems inoperable, operations may continue for up to 30 days provided:
 1. The other two above required leakage detection systems are OPERABLE, and
 2. Appropriate grab samples are obtained and analyzed at least once per 24 hours; otherwise, be in COLD SHUTDOWN within the next 36 hours.
- b. With the containment sump level monitoring system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

- a. Containment atmosphere gaseous and particulate monitoring systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL-TEST at the frequencies specified in Table 4.3-3, and
- b. Containment sump level monitoring system-performance of CHANNEL CALIBRATION TEST at least once per 18 months.

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq 0.10 \mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system $> 0.10 \mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131, be in COLD SHUTDOWN within 36 hours after detection.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

| <u>TYPE OF MEASUREMENT AND ANALYSIS</u> | <u>MINIMUM FREQUENCY</u> |
|---|--|
| 1. Gross Activity Determination | 3 times per 7 days with a maximum time of 72 hours between samples.* |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | a) 1 per 31 days, when- ever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, when- ever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. |

*During Cycle 5 operation, the gross activity of each steam generator shall be determined daily to monitor the extent, if any, of primary to secondary leakage. The following reports shall be provided to the assigned NRC Project Manager:

- a. The baseline primary to secondary leakage rate of each steam generator as determined within two weeks after reaching equilibrium operation in Mode 1; and
- b. Any subsequent incremental increase of ≥ 0.05 gpm in the equilibrium primary-to-secondary leak rate per steam generator be reported verbally within 24 hours with a written followup report within 14 working days.

Whenever possible, all leak rate determinations shall be made at steady state operating conditions.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. DPR-65
NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2
DOCKET NO. 50-336

Introduction

By letters dated January 12 and March 4, 1982 as supplemented by letters of January 22 and February 5, 12 and March 1, 1982, Northeast Nuclear Energy Company (NNECO or the licensee) proposed changes to the Technical Specifications (TS) to allow less than 100 percent inspection of the tubes in each steam generator (SG) and to make additional reports on primary to secondary leakage rates during Cycle 5 operations of Millstone Nuclear Power Station, Unit No. 2 (Millstone-2). The reduced inspection of SG tubes is to be based on the results of sludge analysis and eddy-current testing (ECT) results.

Background

Millstone-2 is currently in the Cycle 5 reload outage. TS 4.4.5.1.2 requires at least a 3% sample of the total number of tubes in all SGs be ECT inspected. NNECO performed ECT inspection of about 1100 and 800 tubes in SG-1 and SG-2, respectively. Entry for these inspections was from the hot leg side taking the probe around the top bend to check for denting indication at the upper support plates or any other tube degradations. Earlier inspection results showed indications (defined in TS as $> 20\%$ tube wall degradation) in some of the central tubes between the tube sheet and the first support plate. The number and severity of indications was such as to require a 100 percent inspection of both SG.

To date, the licensee has performed a 100 percent tube inspection within a sample region of each SG which they believe conservatively bound the area of concern plus additional inspections outside this region primarily in steam generator No. 1. NNECO's request is to limit the required 100 percent inspection to the area of the SG in which sludge pile mapping and the additional inspection indicates the degradation has occurred.

Discussion

Millstone-2 has experienced a moderate amount of denting, particularly at the drilled support plates which has led to the plugging of approximately 800 tubes. Corrective actions taken in 1977 appear to have substantially reduced the rate of denting. These actions included retubing the main turbine condenser with 90-10 cupro-nickle (Cu-Ni), installation of a full flow condensate polishing system, elimination of

hardspot areas ("rimcutting") in the drilled support plates, and improved secondary water chemistry control. Eddy current inspections of the drilled support tube intersections performed during the current outage have indicated some progression of denting since the previous inspection (average progression of 2 mils), but no tubes restricted passage of the standard size eddy current probe at the drilled supports. However, four tubes restricted passage of the eddy current probe at the first egg crate support on the cold leg side and were preventively plugged. Profilometry techniques rather than eddy current testing is the most accurate monitoring method for denting at egg crate supports, due to the tendency of eddy current to underestimate ovalized dents which is the characteristic dent geometry at the egg crate supports. Profilometer inspections have been performed for approximately 800 tubes during the current outage, but the results are still under evaluation by the licensee. Maximum dents measured at the egg crate supports during the previous inspection were approximately 40 mils.

Initial eddy current test sampling revealed a large number of indications in the central bundle regions of both steam generators where sludge has accumulated on the tubesheet. Based on these findings, the licensee defined an inspection in each steam generator encompassing the locations where the initial indications were found, and performed an eddy current inspection of each tube within this boundary (with exception of 54 tubes as discussed below). All indications found as a result of these inspections were located well within the inspection boundary. The inspection boundary bounded the outer most tubes containing indications by a minimum of five tubes.

The licensee reported in its letter dated February 12, 1982 that 33 and 21 tubes in steam generators 1 and 2, respectively, which are located within the inspection boundary defined in the January 12, 1982 letter, were not inspected. These tubes were reportedly inaccessible to the eddy current equipment without extensive equipment modification or personnel radiation exposure. The licensee has justified not inspecting these tubes on the basis that no defective tubes have been identified within at least 5 tubes of any of the uninspected tubes.

Our review indicated that the exclusion of over half of these tubes amounts to a minor adjustment of the inspection boundary consistent with methodology used in the original boundary as developed. The remaining twenty-one tubes are located along row 1 on the hot leg side of both steam generators, immediately adjacent to the open lane between the hot and cold legs.

Tubes located outside the inspection boundary were confined to the peripheral areas of the tube bundle. Sludge mapping indicated the peripheral region of the tube bundle to be essentially free of sludge. However, several hundred

tubes were inspected in the peripheral areas to confirm the absence of pitting in these areas and to monitor the progression of denting. The following table shows the number of tubes inspected and the results.

| | <u>SG-1</u> <u>Hot Leg/Cold Leg</u> | <u>SG-2</u> <u>Hot Leg/Cold Leg</u> |
|--------------------|--|--|
| ECT Inspected | 7200/6900 | 6800/6500 |
| Indications (>20%) | 175/575 | 17/352 |
| Defects (>40%) | 424 | 280 |

All tubes with 40% or greater indications have been plugged in accordance with TS 4.4.5.1.4. This included the 424 tubes in SG-1 and 280 tubes in SG-2, a total of 704 tubes plugged for the pitting problem this outage. This leaves only 151 and 72 tubes, in SG-1 and SG-2, respectively, with known degradation at the beginning of Cycle 5 operation.

The type of plug used is the Westinghouse designed removable plug used previously in other Westinghouse steam generators. NNECO is considering the return of these tubes to service during the next refueling outage by sleeving the defective tubes. This will be reviewed by the staff at a later time.

The eddy current indications occurred at discrete elevations, usually within a narrow (1/4 inch) band for any given tube between 0 and 10 inches above the top of the tubesheet. Sludge mapping established that the indications occurred within the sludge area. Most of the indications occurred on the cold leg side of each steam generator.

- Three tubes containing eddy current indications were removed from SG-1 for laboratory study. Examination of these tubes shows small (< 60 mils) diameter pits as the source of the indications. Similar pitting attack has recently been observed at Indian Point-3 although the magnitude of such defects appears less severe in the Millstone-2 steam generator.

Evaluation of the eddy current data was complicated by the presence of interference signals originating from copper deposits on the tubes. The eddy current test program employed a conventional bobbin probe using multi-frequency techniques (400 and 200 kilo-hertz (kHz) differential, 100 kHz absolute). To verify acceptable sensitivity to these small volume pits, the licensee reinspected twenty tubes using a modified bobbin probe (reduced coil spacing) and a 600 kHz test frequency to minimize signals originating from the outer surface of the tube (e.g. copper signals). Good correlation (i.e. within + 15%) between the conventional probe data and the modified probe data was obtained except for one tube with a 29% difference, the modified probe predicting the larger pit depth of 81%.

Good correlation was also observed between the conventional probe data (i.e. field data) and what was actually observed in the laboratory for one of the pulled tube specimens (81 percent in the field eddy current test vs. 77 percent wastage observed in the lab). Comparative data is not yet available for the other two tube specimens which exhibited field indications of 83% and 28% respectively.

Indications of the type observed during the current outage had not been observed previously. Of these tubes found to contain indications, only 6 tubes had been inspected previously. Eddy current data tapes from previous inspections were reviewed for these tubes with the following results.

| <u>SG Tube Number</u> | <u>August 1980 Outage</u> | <u>December 1981 Outage</u> |
|---------------------------|-------------------------------|---------------------------------|
| 2-137-43 | none | 84% |
| 2-137-45 | none | 99% |
| 1-122-26 | <20% | 21% |
| 1-122-48 | none | 43% |
| 1-114-44 | none | 24% |
| 1-84-100 | <20% | 30% |

The first two tubes were inspected in 1980 by multifrequency while the other 4 tubes were inspected previously by single frequency. These comparisons were obtained by relooking at the 1980 ECT tapes.

The licensee has conducted a test program, described in our meeting on February 15, 1982 and documented by letter dated March 1, 1982, to evaluate the potential effects of pitting on tube integrity. These tests were performed on tube specimens containing simulated pits. The tests included specimens with individual pits and other specimens with multiple arrays of pits, both in a circumferential and axial orientation. These tests included pit diameters ranging to .187 inches which is conservative for pits actually observed at Millstone-2 or Indian Point-3.

Pressure tests by the licensee demonstrate that pits ranging in depth to 92% of the original wall thickness can sustain considerably in excess of the maximum pressure loading associated with a postulated main steam line break (MSLB) with no leakage. When tested to failure, the failures are generally local to the pit (i.e. within the pit) rather than a general or gross failure which could cause large leakage. For severely pitted tubes, the multiplicity of closely spaced pits may degrade tube integrity more than would be indicated by pressure tests of tubes with individual pits since the failure could involve more than one pit. However, even assuming an axial array of four .187 inch

diameter pits, and penetrating 88 percent through wall, with a 10 mil ligament spacing between pits, pressure tests by the licensee have demonstrated burst pressures in excess of postulated MSLB pressures. The licensee also performed a pressure test on one of the pulled tube specimens which exhibited an 83% field ECT indication. This tube exhibited a burst pressure in excess of 9000 psi, which is again indicative of the high residual strength associated with even severely pitted tubes.

Visual inspection of the secondary sides was performed by NNECO from both the top of the bundles and from the lower hand holes. The gap between the upper support plates and the shroud, established by the rim cutting operation in January 1978, was reported to be apparently unchanged since previous inspection. No plate to shroud contact was reported. The licensee inspected for the presence of loose parts such as were found in previous years as a result of cracking of the tube support plates where the rim cut operation was performed. However, no such pieces were found during this inspection.

Evaluation

We have reviewed the licensee's inspection program and results. We find that the licensee has performed adequate inspections to define the areas of the tube bundles affected by the pitting phenomenon. The licensee has inspected essentially all tubes in regions which we believe adequately bound the areas of concern.

Although there were 22 tubes which were not inspected and are located within the inspection boundary and in what can be generally described as the affected area of the steam generator and an area of some sludge accumulation, we believe that it is unlikely that these tubes contain detectable pits based upon our review of spatial distribution of eddy current indications and the fact that there are no indications within a minimum of five tube distance from these uninspected tubes. In addition, considering also the randomness by which pits in this general area tend to occur, the presence of detectable pitting in more than one or two of these tubes is even more unlikely. Finally, even if pitting is present in these tubes, the pitting would be expected to result in only a small leak before tube integrity would become sufficiently degraded to result in a gross tube failure or large leakage (as discussed further below).

The 40% plugging limit implemented during this inspection was developed to ensure adequate tube integrity for generalized forms of corrosion such as uniform thinning of the tube wall or linear defects with uniform wall penetration. In addition, tubes must be inspected at frequent enough intervals to ensure that further progression of corrosion beyond the plugging limit will not excessively degrade tube integrity.

For the case of Millstone-2, we are less than confident that substantial progression of pitting will not occur during the next operating interval. For generalized corrosion involving long linear defects of uniform wall penetration, the allowable wall penetration to ensure no gross failure under postulated MSLB conditions is 78%. To provide a margin of three against burst under normal operating conditions (per Regulatory Guide 1.121), the allowable wall penetration (again assuming generalized corrosion) is 63%. However, the corrosion problem at Millstone-2 involves small diameter pits, which have a substantially smaller effect on tube integrity than more generalized forms of corrosion as has been substantiated by tests. Tube examinations at both Millstone-2 and Indian Point-3 indicate that the pitted tube surface generally would not be expected to exhibit continuous linear flaws of any significant length (say > .75 inches) involving significant average wall penetrations (say > 78%). We believe that the burst test tube specimens which contained a linear array of individual pits, spaced 10 mils apart, over a 3/4 inch length provides a reasonable conservative bound on the potential for individual pits to interact. For this case, the test results submitted by the licensee demonstrate that each of the pits in the array would have to penetrate in excess of 88% through wall to reduce the burst strength of the tubing to less than the MSLB pressure differential. Therefore, considering that the amount of through wall penetration for individual pits will vary randomly, we believe that the further progression of pitting can be expected to result in small leaks before tube integrity becomes sufficiently degraded to result in a gross tube failure during normal or postulated accident conditions. Similarly, leaks would be expected before large numbers of individual pits have penetrated through wall (> 90%) such that significant leakage might result during postulated accident conditions. Allowable limits on primary to secondary leakage (i.e., 0.5 gpm) are specified in the TS 3.4.6.2 beyond which the plant must shut down for steam generator inspections and repairs.

The licensee plans to operate for a full cycle (approximately 340 effective full power days (EFPD)) prior to performing the next steam generator inspection. We do not find data at this time (pending the next steam generator inspection) to make any meaningful estimate of the rate of pitting progression during the upcoming operating cycle. In view of the extensive numbers of tubes affected by pitting, and the potential for significant progression of pitting during the upcoming cycle, we find it necessary to keep ourselves informed of the minimum reasonable increase in primary to secondary leakage. To accomplish this goal, NNECO committed to determine the secondary gross activity and report to the assigned Project Manager the baseline leakage of

each steam generator (determined after two weeks of equilibrium operation) and any subsequent incremental increase of 0.05 or more within 24 hours; followed by a written report within 14 working days. The written report will contain NNECO's evaluation of the increased leakage and justification for continued operation. The staff may, after review of these increased leakage reports, take action as necessary, including ordering a reactor shutdown, to protect the primary system integrity. NNECO's above commitment has been documented by application for TS changes, dated March 4, 1982. The affected TS is Page 3/4 7-8.

These additional reporting requirements apply only to primary to secondary leakage in excess of the baseline leakage to be determined by the end of the second week following restart from the current outage. We believe that any primary to secondary leakage immediately following the current outage would likely be caused by weeping plugs such as has occurred recently at this facility. The staff notes that the above restrictions are in-addition to the current TS requirements which specifies an immediate plant shutdown in the event of primary to secondary leakage exceeding 0.5 gpm.

Based on the above evaluation and the new sampling and reporting requirements, we conclude that the inspection program implemented during the Cycle 5 refueling outage, and thus the requested changes to the TS, are acceptable. In addition, we conclude that operation of the Millstone-2 steam generators during the next operating cycle, subject to the added operating commitments does not represent an undue risk to public health or safety. The staff did request, however, that the licensee commit to submitting the results of the profilimentary inspections at the egg crate supports, and their assessment of the results within 60 days following restart from the current Cycle 5 refueling outage. In addition, we believe NNECO should inspect the 22 tubes identified in their February 12, 1982 letter which were not inspected during the current inspection, but which are located in areas affected by pitting. NNECO has agreed to these two additional requests as documented in their March 4, 1982 letter.

Environmental Consideration

We have determined that the amendment does 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 amendment involves 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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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.

Date: MAR 5 1982

Principal Contributors:
Conrad McCracken
Emmit Murphy
Monte Conner

UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NO. 50-336NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 73 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 the Western Massachusetts Electric Company (the licensee), which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2 (the facility) located in the Town of Waterford, Connecticut. The amendment is effective as of the date of issuance.

The amendment changes the Technical Specifications to allow reactor heatup for Cycle 5 operation with less than 100 percent inspection of the tubes in each steam generator based on the results of the tube inspection program reported in the application and additional reporting requirements on primary-to-secondary steam generator leakage rates. The Cycle 5 reload amendment will be issued under separate cover.

The applications for the amendment comply 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.

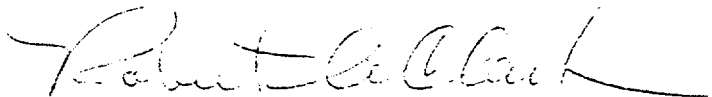
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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated January 12 and March 4, 1982 as modified by letters dated January 22, February 2, 12 and March 1, 1982, (2) Amendment No. 73 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. 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 Licensing.

Dated at Bethesda, Maryland, this 5th day of March, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing